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Docket Nos.: 50-315
50-316

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Donald C. Cook Nuclear Plant Units 1 and 2
TRANSMITTAL OF REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM

Reference:

1) Letter from P-T Kuo, U. S. Nuclear Regulatory Commission (NRC), to M. K. Nazar, Indiana Michigan Power Company (I&M), "Safety Evaluation Report Related to the License Renewal of the Donald C. Cook Nuclear Plant, Units 1 and 2," dated May 29, 2005. Agencywide Documents Access and Management System (ADAMS) Accession No. ML051510015.

2) Regulatory Issue Summary 2011-07, "License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management," dated July 21, 2011. ADAMS Accession No. ML111990086.

3) Letter from J. P. Gebbie, Indiana Michigan Power Company, to U. S. Nuclear Regulatory Commission, "Donald C. Cook Nuclear Plant Units 1 and 2, Docket No. 50-315 and 50-316, Revision to Regulatory Commitments Associated with Application for Renewed Operating Licenses," AEP-NRC-2011-38, dated September 1, 2011. ADAMS Accession No. ML11256A017.

Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP), Units 1 and 2, is submitting the Reactor Vessel Internals (RVI) Aging Management Program (AMP). By Reference 1, the NRC published Safety Evaluation Report (SER) Related to the License Renewal of the CNP. The SER contained a list of commitments made by I&M, specifically, I&M committed to submit the RVIs Plates, Forgings, Welds, and Bolting Program for NRC Staff review and approval three years prior to the period of extended operations. I&M also committed to implement the Cast Austenitic Stainless Steel (CASS) Evaluation Program prior to the period of extended operation. By Reference 3, I&M submitted a revision to the commitment regarding RVIs Plates, Forgings, Welds, and Bolting Program consistent with the guidance contained in Reference 2. The guidance contained in Reference 2 allowed CNP to modify the commitments to reflect a requirement to submit the AMP for Unit 1 no later than October 1, 2012. The due date for Unit 2 remains December 23, 2014.

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NRR

Enclosure 1 to this letter provides the CNP RVI AMP and satisfies the commitments for submitting the Unit 1 and Unit 2 RVIs Plates, Forgings, Welds, and Bolting Program and implementing the CASS Evaluation Program. Enclosure 2 contains a list of commitments made in the RVI AMP.

Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Joel P. Gebbie

Joel P. Gebbie
Site Vice President

DMB/kmh

Enclosures:
1. Donald C. Cook Nuclear Plant Reactor Vessel Internals Aging Management Program
2. List of Regulatory Commitments

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ENCLOSURE 1 TO AEP-NRC-2012-82

DONALD C. COOK NUCLEAR PLANT
REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM

Donald C. Cook Nuclear Plant
Units 1 and 2

Reactor Vessel Internals
Aging Management Program

Donald C. Cook Nuclear Plant
Reactor Vessel Internals Aging Management Program

REVISION APPROVAL SHEET

TITLE: Reactor Vessel Internals Aging Management Program
Donald C. Cook Nuclear Plant, Units 1 and 2

PROGRAM ACCEPTANCE

Prepared: Kevin A. Kalchik, 8/31/12
Kevin A. Kalchik
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Reviewed: Alex M. Olp, 08/31/2012
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The Revision Approval Sheet will be signed and the following Revision Control Sheet shall be completed to provide a record of the revision history each time this document is revised. The signatures above apply only to the changes made in the revision noted. Signatures for superseded revisions are retrievable through Donald C. Cook Nuclear Plant archives.

REVISION CONTROL SHEET

Major changes shall be outlined within the table below. Minor editorial and formatting revisions are not required to be logged.

REVISION	DATE	REVISION SUMMARY
0	9/3/2012	Initial issuance. Prepared: K. Kalchik Reviewed: A. Olp Approved: D. Naughton

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LIST OF ACRONYMS AND ABBREVIATIONS

AMP	Aging Management Program
AMR	Aging Management Review
ARDM	Age Related Degradation Mechanism
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
B&W	Babcock & Wilcox
BMI	Bottom Mounted Instrumentation
BWR	Boiling Water Reactor
CAP	Corrective Actions Program
CASS	Cast Austenitic Stainless Steel
CE	Combustion Engineering
CFR	Code of Federal Regulations
CLB	Current Licensing Basis
CRDM	Control Rod Drive Mechanism
CRGT	Control Rod Guide Tube
CNP	Donald C. Cook Nuclear Plant
EC	Engineering Change
EFPY	Effective Full Power Year
EPRI	Electric Power Research Institute
ET	Electromagnetic Testing (eddy current)
EVT	Enhanced Visual Testing
FMECA	Failure Mode, Effects, and Criticality Analysis
GALL	Generic Aging Lessons Learned
I&E	Inspection and Evaluation
I&M	Indiana & Michigan Power
IASCC	Irradiation Assisted Stress Corrosion Cracking
IGSCC	Inter-Granular Stress Corrosion Cracking
INPO	Institute of Nuclear Power Operations
ISI	In-Service inspection
ISR	Irradiation-enhanced Stress Relaxation
JCO	Justification for Continued Operation
LRA	License Renewal Application
LRSS	Lower Radial Support System
MRP	Materials Reliability Program
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NOS	Nuclear Oversight Section
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OE	Operating Experience
OEM	Original Equipment Manufacturer
PH	Precipitation Hardenable
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group

PWSCC	Primary Water Stress Corrosion Cracking
QA	Quality Assurance
QAPD	Quality Assurance Program Description
RAI	Request for Additional Information
RCS	Reactor Coolant System
RFO	Refueling Outage
RI-FG	Reactor Internals Focus Group
RFO	Re-Fueling Outage
RV	Reactor Vessel
RVI	Reactor Vessel Internals
SCC	Stress Corrosion Cracking
SE	Safety Evaluation
SRP	Standard Review Plan
SS	Stainless Steel
UCP	Upper Core Plate
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Testing
VT	Visual Testing
WCAP	Westinghouse Commercial Atomic Power
WOG	Westinghouse Owners Group

1.0 PURPOSE

The purpose of the Donald C. Cook Nuclear Plant Reactor Vessel Internals Aging Management Program is to manage the effects of aging on reactor vessel internals for the remainder of the operating license of the plant. The program implements guidance from the Electric Power Research Institute provided in MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor internals Inspection and Evaluation Guidelines" (MRP-227-A), and MRP-228, "Materials Reliability Program: Inspection Standard for PWR Internals" (MRP-228) to manage aging effects on the reactor vessel internal components. Existing programs are credited including the American Society of Mechanical Engineers Boiler & Pressure Vessel Code Section XI In-Service Inspection Program, Primary Strategic Water Chemistry Plan, and Incore Instrumentation Thimble Tube Multifrequency Eddy Current Inspections.

The program meets Donald C. Cook Nuclear Plant Nuclear Regulatory Commission commitments for license renewal as described in Appendix A of NUREG-1831, "Safety Evaluation Report Related to the License Renewal of the Donald C. Cook Nuclear Plant, Units 1 and 2" (NUREG-1831). Specifically, NUREG-1831 Appendix A Items 19, 20, and 36, as modified by AEP-NRC-2011-38 in accordance with NRC RIS 11-07 "NRC Regulatory Issue Summary 2011-07 License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management." These commitments require that a Reactor Vessel Internals Aging Management Program be developed in accordance with industry guidance and provided to the Nuclear Regulatory Commission for review and approval.

2.0 DESRIPTION OF CNP REACTOR INTERNALS

Donald C. Cook Nuclear Plant Units 1 and 2 are four-loop Westinghouse reactors in the downflow configuration. A schematic view of typical Westinghouse internals from MRP-227-A is shown in Figure 2-1. Illustrations of components can be found in Appendix E.

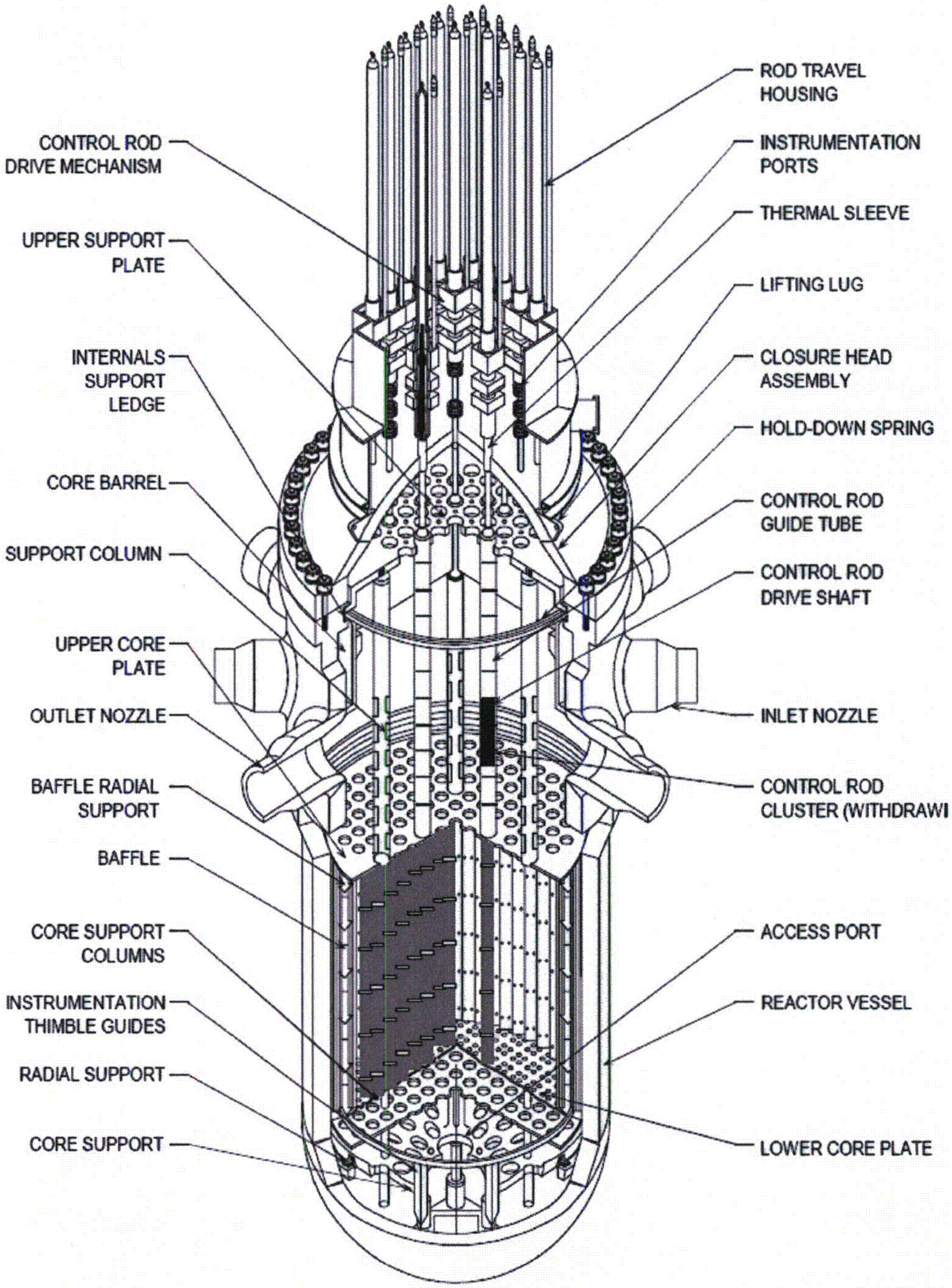

Figure 2-1 Overview of Typical Westinghouse Internals

The following summary of Westinghouse PWR internals is an excerpt from MRP-227-A:

All Westinghouse internals consist of two basic assemblies: an upper internals assembly that is removed during each refueling operation to obtain access to the reactor core and a lower internals assembly that can be removed following a complete core off-load.

The lower internals assembly is supported in the vessel by clamping to a ledge below the vessel-head mating surface and is closely guided at the bottom by radial support/clevis assemblies. The upper internals assembly is clamped at this same ledge by the reactor vessel head. The bottom of the upper internals assembly is closely guided by the core barrel alignment pins of the lower internals assembly.

Upper Internals Assembly

The major sub-assemblies that comprise the upper internals assembly are the: (1) upper core plate (UCP) and fuel alignment pins; (2) upper support column assemblies; (3) control rod guide tube assemblies and flow downcomers; (4) upper plenum; and (5) upper support plate assembly.

During reactor operation, the upper internals assembly is preloaded against the fuel assembly springs and the internals holddown springs by the reactor vessel head pressing down on the outside edge of the upper support plate (USP). The USP acts as the divider between the upper plenum and the reactor vessel head and as a relatively stiff base for the rest of the upper internals. The upper support columns and the guide tubes are attached to the USP. The UCP, in turn, is attached to the upper support columns. The USP assemblies are designated as one of three different designs: (1) a deep beam design, (2) a top hat design, or (3) an inverted top hat design.

The UCP is perforated to permit coolant to pass from the core below into the upper plenum defined by the USP and the UCP. The coolant then exits through the outlet nozzles in the core barrel. The UCP positions and laterally supports the core by fuel alignment pins extending below the plate. The UCP contacts and preloads the fuel assembly springs and thus maintains contact of the fuel assemblies with the lower core plate (LCP) during reactor operation.

The upper support columns vertically position the UCP and are designed to take the uplifting hydraulic flow loads and fuel spring loads on the UCP. The guide tubes are bolted to the USP and pinned at the UCP so they can be easily removed if replacement is desired. The guide tubes are designed to guide the control rods in and out of the fuel assemblies to control power generation. The guide tubes are also slotted in their lower sections to allow coolant exiting from the core to flow into the upper plenum.

The upper instrumentation columns are bolted to the USP. These columns support the thermocouple guide tubes that lead the thermocouples from the reactor head into the upper plenum to just above the UCP.

The UCP alignment pins locate the UCP laterally with respect to the lower internals assembly. The pins must laterally support the UCP so that the plate is free to expand radially and move axially during differential thermal expansions between the upper internals and the core barrel. The UCP alignment pins are the interfacing components between the UCP and the core barrel. The UCP alignment pins are shrunk-fit and welded into the core barrel and the core barrel bearing pad. The gap sizes between the alignment pins and the matching inserts are customized.

The USP, the upper support columns, and the UCP are typically considered core support structures.

Lower Internals Assembly

The reactor core is positioned and supported by the lower internals and upper internals assemblies. The individual fuel assemblies are positioned by fuel alignment pins in the LCP and in the UCP. These pins control the orientation of the core with respect to the lower internals and upper internals assemblies. The lower internals are aligned with the upper internals by the UCP alignment pins and secondarily by the head/vessel alignment pins. The lower internals are orientated to the vessel by the lower radial keys and by the head/vessel alignment pins. Thus, the core is aligned with the vessel by a number of interfacing components.

The lower internals assembly is supported in the vessel by clamping to a ledge below the vessel-head mating surface and closely guided at the bottom by radial support/clevis assemblies. The upper internals assembly is clamped at this same ledge by the reactor vessel head. The bottom of the upper internals assembly is closely guided by the core barrel alignment pins of the lower internals assembly.

The fuel assemblies are supported inside the lower internals assembly on top of the LCP. The LCP is elevated above the lower support forging by support columns and bolted to a ring support attached to the inside diameter of the core barrel. The support columns transmit vertical fuel assembly loads from the LCP to the much thicker lower support forging. The lower support forging is welded to and supported by the core barrel, which transmits vertical loads to the vessel through the core barrel flange.

The functions of the LCP are to position and support the core and provide a metered control of reactor coolant flow into each fuel assembly. The LCP is located near the bottom of the lower support assembly, inside the core barrel, and above the lower support forging.

The function of the lower support forging or casting is to provide support for the core. The lower support forging is attached with a full-penetration weld to the lower end of the core barrel. In this position it can provide uninterrupted support to the core. The core sits directly on the LCP, which is supported by the lower support columns that are attached to and extend above the lower support forging. Some four-loop plants employ a *cast lower support instead of a forging. The functions, loads, and supporting hardware are the same except for dimensions.*

The primary function of the core barrel is to support the core. A large number of components are attached to either the core barrel or the core barrel flange, including the baffle/former assembly, the outlet nozzles, the neutron panel assemblies or thermal shield, the alignment pins that engage the UCP and the LCP, the lower support forging, and the LCP. The radial keys restrain large transverse motions of the core barrel but at the same time allow unrestricted radial and axial thermal expansions.

The baffle and former assembly is made up of vertical plates called baffles and horizontal support plates called formers. The baffle plates are bolted to the formers by the baffle/former bolts, and the formers are attached to the core barrel inside diameter by the barrel/former bolts. The baffle/former assembly forms the interface between the core and the core barrel. The baffles provide a barrier between the core and the former region so that a high concentration of flow in the core region can be maintained. A secondary benefit, although not a requirement of the baffles, is to reduce the neutron flux on the vessel.

Baffle plates are secured to each other at selected corners by edge bolts. In addition, in some installations, corner brackets are installed behind and bolted to the baffle plates.

The function of the core barrel outlet nozzles is to direct the reactor coolant, after it leaves the core, radially outward through the reactor vessel outlet nozzles. The core barrel outlet nozzles are located in the upper portion of the core barrel directly below the flange and are attached to the core barrel, each in line with a vessel outlet nozzle.

Additional neutron shielding of the reactor vessel is provided in the active core region by neutron panels or thermal shields that are attached to the outside of the core barrel. Specimen guides that contain specimens for determining the irradiation effects of the vessel during the life of the plant are attached to the neutron panels/thermal shields.

The flux thimble is a long, slender stainless steel tube that passes from an external seal table, through the bottom mounted nozzle penetration, through the lower internals assembly, and finally extends to the top of the fuel assembly. The flux thimble provides a path for the neutron flux detector into the core and is subjected to reactor coolant pressure and temperature on the outside surface and to atmospheric conditions on the inside. The flux thimble path from the seal table to the bottom mounted nozzles is defined by flux thimble guide tubes, which are part of the primary pressure boundary and not considered to be part of the internals. The bottom-mounted instrumentation (BMI) columns provide a path for the flux thimbles from the bottom of the vessel into the core. The BMI columns align the flux thimble path with instrumentation thimbles in the fuel assembly.

The LCP and the fuel alignment pins, the lower support forging or casting, the lower support columns, the core barrel, the core barrel flange, the radial support keys, the baffle plates, and the former plates are typically classified as core support structures.

2.1 Unit 1 Operating Experience and Records Search

A review of plant records was performed to locate unit specific RVI OE and design changes. Some examples of the results for Unit 1 are discussed in the following sections.

2.1.1 Control Rod Guide Tube and Split Pin Replacement

The original split pins were fabricated from alloy X-750. These pins were replaced with an improved stress design fabricated from alloy X-750 in 1985 (U1C9). Installation was expedited through replacement of control rod guide tube (CRGT) assemblies using spares available from a CNP Unit 2 modification discussed in Section 2.2.1.

2.1.2 Barrel-Former Bolt Inspection and Partial Replacement

A barrel-former bolt was discovered on the lower core plate after defueling in 1994 (U1C14). Inspection was performed in 1995 (U1C15) to determine the origin of the retrieved bolt. All barrel-former bolts were visually inspected and a sample of bolts was mechanically agitated to determine if additional bolts were loose. The two horizontally adjacent bolts to the vacant location were loose. A total of three bolts were replaced with oversized bolts. Replacement efforts required three holes to be machined into the core barrel for tool access. This work was completed in 1997 (U1C16).

2.1.3 Clevis Insert Bolt Degradation

Indications were discovered in the clevis bolts of the lower radial support system while performing the ASME 10-year ISI in 2010 (U1C23). The plant is currently operating based on a JCO provided by the OEM. Development of a repair methodology and associated tooling is under development.

2.2 Unit 2 Operating Experience and Records Search

A review of plant records was performed to locate unit specific RVI OE and design changes. Some examples of the results for Unit 2 are listed in the following sections.

2.2.1 Modification from 15X15 to 17X17 Design

Modifications were made on Unit 2 to accept 17X17 fuel assemblies. Appropriate changes were made to the RVIs at the manufacturer's shop prior to operation. Spare parts generated from the modification, including the 15X15 CRGT assemblies, were retained by I&M. These spare CRGT assemblies were later installed in Unit 1 as described in Section 2.1.1.

2.2.2 Control Rod Guide Tube Split Pin Replacement

The original Unit 2 split pins were fabricated from alloy X-750. A small number of these pins failed and were retrieved from two steam generators in 1985. A JCO was provided by the OEM to operate for the remainder of the fuel cycle with this known degraded condition. The original pins were replaced during the following RFO with an improved stress design fabricated from alloy X-750. This work was completed in 1986 (U2C6).

2.2.3 Control Rod Guide Tube Cap Screw Modification

Each CRGT in Unit 2 has four hold down socket head cap screws fastening it to the support plate. A number of bolts and threaded holes were damaged during the Unit 2 CRGT split pin replacement campaign. Two CRGT hold down socket head cap screws were broken during untorquing, leaving the threaded portion of the cap screws in the tapped support plate holes. Also, the threads of two tapped holes were damaged during the split pin replacement effort. Each damaged location was on a different CRGT assembly. These bolt locations were abandoned as supported by analysis from the OEM. Three high strength bolts were installed at the remaining available locations on these four CRGT assemblies. This work was completed in 1986 (U2C6).

2.2.4 Baffle-Former Bolt Partial Replacement

CNP Unit 2 original baffle-former bolts are internal hex with a cross tack welded lock bar. A number of baffle-former bolts were discovered on the lower core plate in 2010 (U2C19). Visual inspection revealed 18 failed bolts ranging from broken or missing lock bars to broken or missing bolt heads in a local area on the large south baffle plate. Bolts with visual indications were replaced. Replacement was expanded to bolts in adjacent rows and columns in the plate to bound the edge of the local degradation. Bolt samples were removed from the other three large baffle plates and inspected to ensure degradation was not occurring at symmetric locations. A total of 52 bolts were replaced with two locations left vacant.

3.0 EXISTING PROGRAMS AND ACTIVITIES

There are a number of programs that support the CNP RVI AMP. These existing and ongoing programs are an integral part of the aging management strategy at CNP.

3.1 ASME Section XI ISI Program

The ASME Section XI program monitors for aging effects in reactor vessel internals through periodic inspections. Results are dispositioned in accordance with the appropriate acceptance criteria provided in the code. This program has been effective at identifying and managing aging in the reactor vessel internals.

3.2 Primary Strategic Water Chemistry Plan

The Primary Strategic Water Chemistry Plan is used to control water chemistry to minimize or eliminate material degradation due to contaminants. This program monitors chemistry and maintains concentrations within the system-specific tolerance. The program follows the guidance provided in the EPRI PWR Water Chemistry Guidelines. This program has been effective for controlling water chemistry to minimize material degradation.

3.3 Thimble Tube Multifrequency Eddy Current Inspection

The Incore Instrumentation Thimble Tube Multifrequency Eddy Current Inspection program periodically inspects for thimble tube wear in accordance with NRC Bulletin 88-09. This program has been effective in identifying loss of material due to wear prior to leakage. This has allowed pre-emptive corrective actions to maintain proactive management of the thimble tubes.

3.4 Industry Involvement

CNP participates in industry activities including the Pressurized Water Reactor owner's Group and the Electric Power Research Institute Materials Reliability Program. Participation includes attending meetings, providing input to work products, and implementing work products as applicable and appropriate for CNP.

4.0 INSPECTION & EVALUATION GUIDELINES

I&M's strategy for managing aging effects of reactor vessel internals at CNP includes performing augmented inspections as described in guidance provided by the EPRI MRP. The MRP inspection & evaluation guidelines for managing the effects of aging on PWR internals are documented in MRP-227-A. These guidelines do not reduce, alter, or otherwise affect current ASME B&PV Code Section XI or plant-specific licensing inservice inspection requirements. The MRP developed the companion document MRP-228 which contains requirements specific to the inspection methodologies involved as well as requirements for qualification of the NDE systems used to perform those inspections.

All PWR internals were placed into four functional groups. The following is an excerpt from MRP-227-A, Section 3.3.1.

- ***Primary:*** *those PWR internals that are highly susceptible to the effects of at least one of the eight aging mechanisms were placed in the Primary group. The aging management requirements that are needed to ensure functionality of Primary components are described in these I&E guidelines. The Primary group also includes components which have shown a degree of tolerance to a specific aging degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible.*

- ***Expansion:*** *those PWR internals that are highly or moderately susceptible to the effects of at least one of the eight aging mechanisms, but for which functionality assessment has shown a degree of tolerance to those effects, were placed in the Expansion group. The schedule for implementation of aging management requirements for Expansion components will depend on the findings from the examinations of the Primary components at individual plants.*

- ***Existing Programs:*** *those PWR internals that are susceptible to the effects of at least one of the eight aging mechanisms and for which generic and plant-specific existing AMP elements are capable of managing those effects, were placed in the Existing Programs group.*

- ***No Additional Measures:*** *those PWR internals for which the effects of all eight aging mechanisms are below the screening criteria were placed in the No Additional Measures group. Additional components were placed in the No Additional Measures group as a result of FMECA and the functionality assessment. No further action is required by these guidelines for managing the aging of the No Additional Measures components.*

The categorization and analysis processes described herein are not intended to supersede any ASME B&PV Code Section XI [2] requirements. Any components that are classified as core support structures as defined in ASME B&PV Code Section XI IWB 2500 IWA 9000, and listed in Table IWB 2500-1. Category B-N-3 [2] have requirements that remain in effect and may only be altered as allowed by 10CFR50.55a [4].

4.1 Guideline Background

The first EPRI MRP guidance for PWR RVI AMPs was published in December 2008 as MRP-227, Revision 0. EPRI submitted the report for NRC staff review and approval in January 2009. The NRC issued the final SE, Revision 0, for MRP-227, Revision 0 in June 2011. Revision 1 to the SE on MRP-227, Revision 0 was issued in December 2011 and included in MRP-227-A published in December 2011.

4.2 MRP-227-A Applicability to CNP

There are three general assumptions used in the MRP-227-A.

4.2.1 General Assumption 1

The following is the first general assumption from Section 2.4 of MRP-227-A:

> *30 years of operation with high leakage core loading patterns (fresh fuel assemblies loaded in peripheral locations) followed by implementation of a low-leakage fuel management strategy for the remaining 30 years of operation.*

Both CNP units changed from a high leakage to a low leakage core pattern prior to 30 years of operation. It is more conservative to operate with a low leakage core pattern than a high leakage core pattern. Therefore, both CNP units are bounded by this assumption.

4.2.2 General Assumption 2

The following is the second general assumption from Section 2.4 of MRP-227-A:

> *Base load operation, i.e., typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule.*

Both CNP units are base load plants which operate at a fixed power level and do not vary power on a calendar or load demand schedule. Therefore, both CNP units are bounded by this assumption.

4.2.3 General Assumption 3

The following is the third general assumption from Section 2.4 of MRP-227-A:

No design changes beyond those identified in general industry guidance or recommended by the original vendors.

All U.S. PWR operating plants met this assumption as of May 2007 for the three designs identified in MRP-227-A. Each CNP unit is discussed individually in the following sections.

4.2.3.1 Unit 1 Applicability

No modifications have been made to CNP Unit 1 RVIs since May 2007. Therefore CNP Unit 1 is bounded by this assumption.

Degraded bolts and a degraded dowel pin were discovered in the LRSS clevis inserts in the RV during the 2010 RFO (U1C23). The unit is operating with these degraded bolts and degraded dowel pin on an interim analysis while I&M prepares for a repair. Section 2.1.3 discusses this topic in further detail. I&M has engaged the OEM for analysis in support of a repair. CNP Unit 1 will continue to be bounded by this assumption following repair.

4.2.3.2 Unit 2 Applicability

Baffle-former bolt degradation was observed and repaired during the 2010 RFO (U2C19). Section 2.2.4 discusses this topic in further detail. Repair was performed by the OEM. No other modifications have been made to the CNP Unit 2 RVIs. Therefore, CNP Unit 2 is bounded by this assumption.

4.3 NEI 03-08 Guidance in MRP-227-A

There are one "Mandatory", five "Needed", and zero "Good Practice" elements identified in MRP-227-A under the NEI-03-08 implementation protocol. These elements are discussed in the following sections.

4.3.1 NEI 03-08 Mandatory

The following is the NEI 03-08 "mandatory" element from Section 7.2 of MRP-227-A:

> *Each commercial U.S. PWR unit shall develop and document a program for management of aging of reactor internal components within thirty-six months following issuance of MRP-227-Rev. 0 (that is, no later than December 31, 2011).*

This mandatory element requires that a program for management of aging of reactor internal components is developed by December 31, 2011. WCAP-17300, "Reactor Vessel Internals Program Plan for Aging Management of Reactor Internals at D.C. Cook Nuclear Plant Unit 1" and WCAP-17301, "Reactor Vessel Internals Program Plan for Aging Management of Reactor Internals at D.C. Cook Nuclear Plant Unit 2" were completed in February 2011. These documents are superseded by this document. This element is fulfilled for both CNP units.

4.3.2 NEI 03-08 Needed

There are five NEI 03-08 "needed" elements in MRP-227-A. These are addressed in the following sections.

4.3.2.1 Needed Element 1

The following is the NEI 03-08 "needed" element from Section 7.3 of MRP-227-A:

> *Each commercial U.S. PWR unit shall implement Tables 4-1 through 4-9 and Tables 5-1 through 5-3 for the applicable design within twenty-four months following issuance of MRP-227-A.*

MRP-227-A was issued in December 2011 making implementation of the applicable tables needed by December 2013. The applicable Westinghouse tables contained in MRP-227-A are Table 4-3 for primary components, Table 4-6 for expansion components, Table 4-9 for existing programs, and Table 5-3 for acceptance and expansion criteria. These tables are included as Appendix A, Appendix B, Appendix C, and Appendix D respectively. The CNP RVI AMP implements these tables. This element is fulfilled for both CNP units.

4.3.2.2 Needed Element 2

The following is the NEI 03-08 "needed" element from Section 7.4 of MRP-227-A:

> *Examinations specified in these guidelines shall be conducted in accordance with the Inspection Standard (MRP-228).*

MRP-228 is the companion document to MRP-227-A. Internals examinations conducted as specified in MRP-227-A will be in accordance with MRP-228. This element is fulfilled for both CNP units.

4.3.2.3 *Needed Element 3*

The following is the NEI 03-08 "needed" element from Section 7.5 of MRP-227-A:

> *Examination results that do not meet the examination acceptance criteria defined in Section 5 of these guidelines shall be recorded and entered in the plant corrective action program and dispositioned.*

Conditions are documented and dispositioned in accordance with the Corrective Action Program. This element is fulfilled for both CNP units.

4.3.2.4 *Needed Element 4*

The following is the NEI 03-08 "needed" element from Section 7.6 of MRP-227-A:

> *Each commercial U.S. PWR unit shall provide a summary report of all inspections and monitoring, items requiring evaluation, and new repairs to the MRP Program Manager within 120 days of the completion of an outage during which PWR internals within the scope of MRP-227 are examined.*

I&M will provide a summary report of all inspections and monitoring, items requiring evaluation, and new repairs within the scope of MRP-227-A to the MRP Program Manager within 120 days of the completion of an outage during which CNP PWR internals within the scope of MRP-227-A are examined. This element is fulfilled for both CNP units.

4.3.2.5 *Needed Element 5*

The following is the NEI 03-08 "needed" element from Section 7.7 of MRP-227-A:

> *If an engineering evaluation is used to disposition an examination result that does not meet the examination acceptance criteria in Section 5, this engineering evaluation shall be conducted in accordance with a NRC-approved evaluation methodology.*

Inspection results from MRP-227-A inspections that do not meet the acceptance criteria will be dispositioned in accordance with an NRC approved methodology, or the methodology will be submitted for NRC approval prior to implementation. This element is fulfilled for both CNP units.

4.4 Safety Evaluation Report Conditions and Limitations

There are a number of conditions and limitations as described in "Revision 1 to the Safety Evaluation by the Office of Nuclear Reactor Regulation Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 0) Project No. 669." (MRP-227 SE). These include seven Topical Report Conditions and eight Applicant/Licensee Action Items.

4.4.1 Topical Report Conditions

The topical report conditions contained in the MRP-227 SE were incorporated into MRP-227-A. The CNP RVI AMP is consistent with MRP-227-A. Therefore all topical report conditions are fulfilled for both CNP units.

4.4.2 Applicant/Licensee Action Items

The Applicant/Licensee Action Items contained in Revision 1 to the SE on MRP-227, Revision 0 are discussed in the following sections.

4.4.2.1 Applicant/Licensee Action Item 1 (MRP-227 SE Sections 3.2.5.1 and 4.2.1)

"Applicability of FMECA and Functionality Analysis Assumptions" from Section 4.2.1 of the MRP-227 SE:

> *As addressed in Section 3.2.5.1 of this SE, each applicant/licensee is responsible for assessing its plant's design and operating history and demonstrating that the approved version of MRP-227 is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the*

> *FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227.*

This Action Item addresses the applicability of the FMECA and functionality analysis assumptions made in the development of MRP-227-A to individual facilities. I&M is participating in PWROG project PA-MSC-0938, "Support for Applicant Action Items 1, 2, and 7 from the Final Safety Evaluation on MRP-227, Revision 0" to address this item. The results of the evaluation will be provided to the NRC prior to the period of extended operation for each unit.

4.4.2.2 Applicant/Licensee Action Item 2 (MRP-227 SE Sections 3.2.5.2 and 4.2.2)

"PWR Vessel Internal Components Within the Scope of License Renewal" from Section 4.2.2 of the MRP-227 SE:

> *As discussed in Section 3.2.5.2 of this SE, consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which RVI components are within the scope of LR for its facility. Applicants/licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the RVI components that are within the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227, as modified by this SE, when submitting its plant-specific AMP. The AMP shall provide assurance that the effects of aging on the missing component(s) will be managed for the period of extended operation.*

This Action Item requires the licensee to verify that all the RVI components within the scope for license renewal at that facility have been considered in applicable documents in development of MRP-227-A. I&M is participating in PWROG project PA-MSC-0938, "Support for Applicant Action Items 1, 2, and 7 from the Final Safety Evaluation on MRP-227, Revision 0" to address this item. The results of the evaluation will be provided to the NRC prior to the period of extended operation for each unit.

4.4.2.3 Applicant/Licensee Action Item 3 (MRP-227 SE Sections 3.2.5.3 and 4.2.3)

"Evaluation of the Adequacy of Plant-Specific Existing Programs" from Section 4.2.3 of the MRP-227 SE:

> *As addressed in Section 3.2.5.3 in this SE, applicants/licensees of CE and Westinghouse are required to perform plant-specific analysis either to justify the acceptability of an applicant's/licensee's existing programs, or to identify changes to the programs that*

> *should be implemented to manage the aging of these components for the period of extended operation. The results of this plant-specific analyses and a description of the plant-specific programs being relied on to manage aging of these components shall be submitted as part of the applicant's/licensee's AMP application. The CE and Westinghouse components identified for this type of plant-specific evaluation include: CE thermal shield positioning pins and CE in-core instrumentation thimble tubes (Section 4.3.2 in MRP-227), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227).*

CNP Unit 1 and Unit 2 both have X-750 split pins. Project requests have been initiated to investigate split pin replacement for each unit. I&M will provide the NRC with the strategy for managing split pins prior to the period of extended operation for each unit.

4.4.2.4 Applicant/Licensee Action Item 4 (MRP-227 SE Sections 3.2.5.4 and 4.2.4)

"B&W Core Support Structure Upper Flange Stress Relief" from Section 4.2.4 of the MRP-227 SE:

> *As discussed in Section 3.2.5.4 of this SE, the B&W applicants/licensees shall confirm that the core support structure upper flange weld was stress relieved during the original fabrication of the Reactor Pressure Vessel in order to confirm the applicability of MRP-227, as approved by the NRC, to their facility. If the upper flange weld has not been stress relieved, then this component shall be inspected as a "Primary" inspection category component. If necessary, the examination methods and frequency for non-stress relieved B&W core support structure upper flange welds shall be consistent with the recommendations in MRP-227, as approved by the NRC, for the Westinghouse and CE upper core support barrel welds. The examination coverage for this B&W flange weld shall conform to the staff's imposed criteria as described in Sections 3.3.1 and 4.3.1 of this SE. The applicant's/licensee's resolution of this plant-specific action item shall be submitted to the NRC for review and approval.*

This item is specific to the Babcock & Wilcox designed plant and it is not applicable to CNP. No action is required.

4.4.2.5 Applicant/Licensee Action Item 5 (MRP-227 SE Sections 3.3.5 and 4.2.5)

"Application of Physical Measurements as part of I&E Guidelines for B&W, CE, and Westinghouse RVI Components" from Section 4.2.5 of the MRP-227 SE:

> *As addressed in Section 3.3.5 in this SE, applicants/licensees shall identify plant-specific acceptance criteria to be applied when performing the physical measurements required by the NRC-approved version of MRP-227 for loss of compressibility for Westinghouse hold down springs, and for distortion in the gap between the top and bottom core shroud segments in CE units with core barrel shrouds assembled in two vertical sections. The applicant/licensee shall include its proposed acceptance criteria and an explanation of how the proposed acceptance criteria are consistent with the plants' licensing basis and*

> *the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation during the period of extended operation as part of their submittal to apply the approved version of MRP-227.*

CNP Unit 1 and Unit 2 both have 304 SS hold down springs. MRP-227-A guidance includes physical measurement of 304 SS hold down springs. This action item requires acceptance criteria to be provided to the NRC. CNP plant specific acceptance criteria will be developed and submitted to the NRC prior to the first required physical measurement. The hold down springs will be replaced if acceptance criteria are not developed in lieu of performing the first required physical measurement.

4.4.2.6 Applicant/Licensee Action Item 6 (MRP-227 SE Sections 3.3.6 and 4.2.6)

"Evaluation of Inaccessible B&W Components" from Section 4.2.6 of the MRP-227 SE:

> *As addressed in Section 3.3.6 in this SE, MRP-227 does not propose to inspect the following inaccessible components: the B&W core barrel cylinders (including vertical and circumferential seam welds), B&W former plates, B&W external baffle-to-baffle bolts and their locking devices, B&W core barrel-to-former bolts and their locking devices, and B&W core barrel assembly internal baffle-to-baffle bolts. The MRP also identified that although the B&W core barrel assembly internal baffle-to-baffle bolts are accessible, the bolts are non-inspectable using currently available examination techniques.*
>
> *Applicants/licensees shall justify the acceptability of these components for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the components. As part of their application to implement the approved version of MRP-227, applicants/licensees shall provide their justification for the continued operability of each of the inaccessible components and, if necessary, provide their plan for the replacement of the components for NRC review and approval.*

This item is specific to the Babcock & Wilcox designed plant and it is not applicable to CNP. No action is required.

4.4.2.7 Applicant/Licensee Action Item 7 (MRP-227 SE Sections 3.3.7 and 4.2.7)

Section 4.2.7, "Plant-Specific Evaluation of CASS Materials" from the MRP-227 SE:

> *As discussed in Section 3.3.7 of this SE, the applicants/licensees of B&W, CE, and Westinghouse reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W IMI guide tube assembly spiders and CRGT spacer castings, CE lower support columns, and Westinghouse lower support column bodies will maintain their functionality during the period of extended operation or for additional RVI components that may be fabricated from CASS, martensitic stainless steel or precipitation hardened stainless steel materials. These analyses shall also consider the*

> *possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The requirement may not apply to components that were previously evaluated as not requiring aging management during development of MRP-227. That is, the requirement would apply to components fabricated from susceptible materials for which an individual licensee has determined aging management is required, for example during their review performed in accordance with Applicant/Licensee Action Item 2. The plant-specific analysis shall be consistent with the plant's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. The applicant/licensee shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227.*

A plant specific evaluation of RVI CASS materials is required in this Action Item. I&M is participating in PWROG project PA-MSC-0938, "Support for Applicant Action Items 1, 2, and 7 from the Final Safety Evaluation on MRP-227, Revision 0" to address this item. The results of the evaluation will be provided to the NRC prior to the period of extended operation for each unit.

4.4.2.8 Applicant/Licensee Action Item 8 (MRP-227 SE Sections 3.5.1 and 4.2.8)

Section 3.5.1, "Submittal of Information for Staff Review and Approval" from the MRP-227 SE:

> *In addition to the implementation of MRP-227 in accordance with NEI 03-08, applicants/licensees whose licensing basis contains a commitment to submit a PWR RVI AMP and/or inspection program shall also make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE. An applicant's/licensee's application to implement MRP-227, as amended by this SE shall include the following items (1) and (2). Applicants who submit applications for LR after the issuance of this SE shall, in accordance with the NUREG-1801, Revision 2, submit the information provided in the following items (1) through (5) for staff review and approval.*

This Action Item includes five parts. However, parts 3-5 are only applicable to licensees who submit license renewal applications after the issuance of the MRP-227 SE. I&M submitted CNP LRA in October 2003, the MRP-227 SE was issued in December 2011. Therefore, parts 3-5 are not applicable to CNP. Action Item 8, parts 1 and 2 are discussed in the following sections.

4.4.2.8.1 GALL Revision 2 Requirement

Section 3.5.1, Part 1 from the MRP-227 SE:

> *An AMP for the facility that addresses the 10 program elements as defined in NUREG-1801, Revision 2, AMP XI.M16A.*

The CNP RVI AMP addresses the ten program elements as defined in NUREG-1801, Revision 2, AMP XI.M16A. This item is addressed in detail in Section 5.0 of this document. Therefore, part 1 of this Action Item is fulfilled for both units.

4.4.2.8.2 RVI AMP Submittal Requirements

Section 3.5.1, Part 2 from the MRP-227 SE:

> *To ensure the MRP-227 program and the plant-specific action items will be carried out by applicants/licensees, applicants/licensees are to submit an inspection plan which addresses the identified plant-specific action items for staff review and approval consistent with the licensing basis for the plant. If an applicant/licensee plans to implement an AMP which deviates from the guidance provided in MRP-227, as approved by the NRC, the applicant/licensee shall identify where their program deviates from the recommendations of MRP-227, as approved by the NRC, and shall provide a justification for any deviation which includes a consideration of how the deviation affects both "Primary" and "Expansion" inspection category components.*

The CNP RVI AMP addresses the plant-specific action items. The CNP RVI AMP does not deviate from MRP-227-A. Therefore, part 2 of this Action Item is fulfilled for both units upon submittal of the CNP RVI AMP to the NRC for review and approval.

5.0 PROGRAM ATTRIBUTE EVALUATION

The CNP RVI AMP addresses the 10 program elements as defined in NUREG-1801, Revision 2, Section XI.M16A. This is in accordance with Applicant/Licensee Action Item 8, part 1 from the MRP-227 SE.

The following is the program description from NUREG-1801, Revision 2, Section XI.M16A:

> ***Program Description***
> *This program relies on implementation of the Electric Power Research Institute (EPRI) Report No. 1016596 (MRP-227) and EPRI Report No. 1016609 (MRP-228) to manage the aging effects on the reactor vessel internal (RVI) components.*
>
> *This program is used to manage the effects of age-related degradation mechanisms that are applicable in general to the PWR RVI components at the facility. These aging effects include (a) various forms of cracking, including stress corrosion cracking (SCC), which also encompasses primary water stress corrosion cracking (PWSCC), irradiation assisted stress corrosion cracking (IASCC), or cracking due to fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness due to either thermal aging or neutron irradiation embrittlement; (d) changes in dimension due to void swelling; and (e) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep.*
>
> *The program applies the guidance in MRP-227 for inspecting, evaluating, and, if applicable, dispositioning non-conforming RVI components at the facility. The program conforms to the definition of a sampling-based condition monitoring program, as defined by the Branch Technical Position RSLB-1, with periodic examinations and other inspections of highly-affected internals locations. These examinations provide reasonable assurance that the effects of age-related degradation mechanisms will be managed during the period of extended operation. The program includes expanding periodic examinations and other inspections if the extent of the degradation effects exceeds the expected levels.*
>
> *The MRP-227 guidance for selecting RVI components for inclusion in the inspection sample is based on a four step ranking process. Through this process, the reactor internals for all three PWR designs were assigned to one of the following four groups: Primary, Expansion, Existing Programs, and No Additional Measures components. Definitions of each group are provided in GALL Chapter IX.B.*
>
> *The result of this four-step sample selection process is a set of Primary Internals Component locations for each of the three plant designs that are expected to show the leading indications of the degradation effects, with another set of Expansion Internals Component locations that are specified to expand the sample should the indications be more severe than anticipated. The degradation effects in a third set of internals locations are deemed to be adequately managed by Existing Programs, such as ASME Code, Section XI,[11] Examination Category B-N-3 examinations of core support structures. A*

fourth set of internals locations are deemed to require no additional measures. As a result, the program typically identifies 5 to 15% of the RVI locations as Primary Component locations for inspections, with another 7 to 10% of the RVI locations to be inspected as Expansion Components, as warranted by the evaluation of the inspection results. Another 5 to 15% of the internals locations are covered by Existing Programs, with the remainder requiring no additional measures. This process thus uses appropriate component functionality criteria, age related degradation susceptibility criteria, and failure consequence criteria to identify the components that will be inspected under the program in a manner that conforms to the sampling criteria for sampling-based condition monitoring programs in Section A.1.2.3.4 of NRC Branch Position RLSB-1. Consequently, the sample selection process is adequate to assure that the intended function(s) of the PWR reactor internal components are maintained during the period of extended operation.

The program's use of visual examination methods in MRP-227 for detection of relevant conditions (and the absence of relevant conditions as a visual examination acceptance criterion) is consistent with the ASME Code, Section XI rules for visual examination. However, the program's adoption of the MRP-227 guidance for visual examinations goes beyond the ASME Code, Section XI visual examination criteria because additional guidance is incorporated into MRP-227 to clarify how the particular visual examination methods will be used to detect relevant conditions and describes in more detail how the visual techniques relate to the specific RVI components and how to detect their applicable age-related degradation effects.

The technical basis for detecting relevant conditions using volumetric ultrasonic testing (UT) inspection techniques can be found in MRP-228, where the review of existing bolting UT examination technical justifications has demonstrated the indication detection capability of at least two vendors, and where vendor technical justification is a requirement prior to any additional bolting examinations. Specifically, the capability of program's UT volumetric methods to detect loss of integrity of PWR internals bolts, pins, and fasteners, such as baffle-former bolting in B&W and Westinghouse units, has been well demonstrated by operating experience. In addition, the program's adoption of the MRP-227 guidance and process incorporates the UT criteria in MRP-228, which calls for the technical justifications that are needed for volumetric examination method demonstrations, required by the ASME Code, Section V.

The program also includes future industry operating experience as incorporated in periodic revisions to MRP-227. The program thus provides reasonable assurance for the long-term integrity and safe operation of reactor internals in all commercial operating U.S. PWR nuclear power plants.

Age-related degradation in the reactor internals is managed through an integrated program. Specific features of the integrated program are listed in the following ten program elements. Degradation due to changes in material properties (e.g., loss of fracture toughness) was considered in the determination of inspection recommendations and is managed by the requirement to use appropriately degraded properties in the *evaluation of identified defects. The integrated program is implemented by the applicant through an inspection plan that is submitted to the NRC for review and approval with the application for license renewal.*

[11] Refer to the GALL Report, Chapter I, for applicability of various editions of the ASME Code, Section XI.

5.1 Scope of Program

The following is NUREG-1801, Revision 2, Section XI.M16A, Evaluation and Technical Basis Element 1:

The scope of the program includes all RVI components at the Donald C. Cook Nuclear Plant Unit1 and Unit 2, which are built to a Westinghouse NSSS design. The scope of the program applies the methodology and guidance in the most recently NRC endorsed version of MRP-227, which provides augmented inspection and flaw evaluation methodology for assuring the functional integrity of safety-related internals in commercial operating U.S. PWR nuclear power plants designed by B&W, CE, and Westinghouse. The scope of components considered for inspection under MRP-227 guidance includes core support structures (typically denoted as Examination Category B-N-3 by the ASME Code, Section XI), those RVI components that serve an intended license renewal safety function pursuant to criteria in 10 CFR 54.4(a)(1), and other RVI components whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii). The scope of the program does not include consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation, because these components are not typically within the scope of the components that are required to be subject to an aging management review (AMR), as defined by the criteria set in 10 CFR 54.21(a)(1). The scope of the program also does not include welded attachments to the internal surface of the reactor vessel because these components are considered to be ASME Code Class 1 appurtenances to the reactor vessel and are adequately managed in accordance with an applicant's AMP that corresponds to GALL AMP XI.M1, "ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD."

The scope of the program includes the response bases to applicable license renewal applicant action items (LRAAIs) on the MRP-227 methodology, and any additional programs, actions, or activities that are discussed in these LRAAI responses and credited for aging management of the applicant's RVI components. The LRAAIs are identified in the staff's safety evaluation on MRP-227 and include applicable action items on meeting those assumptions that formed the basis of the MRP's augmented inspection and flaw evaluation methodology (as discussed in Section 2.4 of MRP-227), and NSSS vendor-specific or plant-specific LRAAIs as well. The responses to the LRAAIs on MRP-227 are provided in Appendix C of the LRA.

The guidance in MRP-227 specifies applicability limitations to base-loaded plants and the fuel loading management assumptions upon which the functionality analyses were

> *based. These limitations and assumptions require a determination of applicability by the applicant for each reactor and are covered in Section 2.4 of MRP-227.*

A description of CNP RVIs is provided in Section 2.0. The scope of the CNP RVI AMP applies the methodology and guidance in MRP-227-A. The program does not consider consumable items or welded attachments to the internal surface of the reactor vessel. The applicable licensee action items from the MRP-227 SE are addressed in Section 4.4. The licensee action item responses are in this document rather than the LRA because the LRA was issued prior to the MRP-227 SER. Determination of the applicability of CNP RVIs to the applicability limitations identified in MRP-227-A is addressed in Section 4.2.

The CNP RVI AMP scope is consistent with NUREG-1801, Revision 2, Section XI.M16A.

5.2 Preventive Actions

The following is NUREG-1801, Revision 2, Section XI.M16A, Evaluation and Technical Basis Element 2:

> *The guidance in MRP-227 relies on PWR water chemistry control to prevent or mitigate aging effects that can be induced by corrosive aging mechanisms (e.g., loss of material induced by general, pitting corrosion, crevice corrosion, or stress corrosion cracking or any of its forms [SCC, PWSCC, or IASCC]). Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program. The program description, evaluation, and technical basis of water chemistry are presented in GALL AMP XI.M2, "Water Chemistry."*

The CNP Primary Water Chemistry Plan is consistent with NUREG-1801, Revision 0, Section XI.M2. Further details on this program can be found in Section 3.2.

The CNP RVI AMP preventive actions are consistent with NUREG-1801, Revision 2, Section XI.M16.

5.3 Parameters Monitored/Inspected

The following is NUREG-1801, Revision 2, Section XI.M16A, Evaluation and Technical Basis Element 3:

> *The program manages the following age-related degradation effects and mechanisms that are applicable in general to the RVI components at the facility: (a) cracking induced by SCC, PWSCC, IASCC, or fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness induced by either thermal aging or neutron irradiation embrittlement; (d) changes in dimension due to void swelling and irradiation growth, distortion, or deflection; and (e) loss of preload caused by thermal and irradiation enhanced stress relaxation or creep. For the management of cracking, the program monitors for evidence of surface breaking linear discontinuities if a visual inspection technique is used as the non-destruction examination (NDE) method, or for*

relevant flaw presentation signals if a volumetric UT method is used as the NDE method. For the management of loss of material, the program monitors for gross or abnormal surface conditions that may be indicative of loss of material occurring in the components. For the management of loss of preload, the program monitors for gross surface conditions that may be indicative of loosening in applicable bolted, fastened, keyed, or pinned connections. The program does not directly monitor for loss of fracture toughness that is induced by thermal aging or neutron irradiation embrittlement, or by void swelling and irradiation growth; instead, the impact of loss of fracture toughness on component integrity is indirectly managed by using visual or volumetric examination techniques to monitor for cracking in the components and by applying applicable reduced fracture toughness properties in the flaw evaluations if cracking is detected in the components and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation under the MRP-227 guidance or ASME Code, Section XI requirements. The program uses physical measurements to monitor for any dimensional changes due to void swelling, irradiation growth, distortion, or deflection.

Specifically, the program implements the parameters monitored/inspected criteria for Westinghouse designed Primary Components in Table 4-3 of MRP-227. Additionally, the program implements the parameters monitored/inspected criteria for Westinghouse designed Expansion Components in Table 4-6 of MRP-227. The parameters monitored/inspected for Existing Program Components follow the bases for referenced Existing Programs, such as the requirements for ASME Code Class RVI components in ASME Code, Section XI, Table IWB-2500-1, Examination Categories B-N-3, as implemented through the applicant's ASME Code, Section XI program, or the recommended program for inspecting Westinghouse designed flux thimble tubes in GALL AMP XI.M37, "Flux Thimble Tube Inspection." No inspections, except for those specified in ASME Code, Section XI, are required for components that are identified as requiring "No Additional Measures," in accordance with the analyses reported in MRP-227.

The CNP RVI AMP manages age-related degradation effects including SCC, IASCC, wear, fatigue, thermal aging embrittlement, irradiation embrittlement, void swelling and irradiation growth, and thermal and irradiation-enhanced stress relaxation or irradiation-enhanced creep. These effects are monitored using visual examination, surface examination, volumetric examination, and physical measurements. The program implements Table 4-3 and Table 4-6 from MRP-227-A which are included as Appendix A and Appendix B, respectively, in this document. The program credits the ASME Section XI ISI program described in Section 3.1 which is consistent with NUREG-1801, Revision 0, XI.M1. The program also credits the thimble tube inspection program which is consistent with the intent of the 10 GALL elements.

The CNP RVI AMP parameters monitored/inspected are consistent with NUREG-1801, Revision 2, Section XI.M16A.

5.4 Detection of Aging Effects

The following is NUREG-1801, Revision 2, Section XI.M16A, Evaluation and Technical Basis Element 4:

> *The detection of aging effects is covered in two places: (a) the guidance in Section 4 of MRP-227 provides an introductory discussion and justification of the examination methods selected for detecting the aging effects of interest; and (b) standards for examination methods, procedures, and personnel are provided in a companion document, MRP-228. In all cases, well-established methods were selected. These methods include volumetric UT examination methods for detecting flaws in bolting, physical measurements for detecting changes in dimension, and various visual (VT-3, VT-1, and EVT-1) examinations for detecting effects ranging from general conditions to detection and sizing of surface-breaking discontinuities. Surface examinations may also be used as an alternative to visual examinations for detection and sizing of surface-breaking discontinuities.*
>
> *Cracking caused by SCC, IASCC, and fatigue is monitored/inspected by either VT-1 or EVT-1 examination (for internals other than bolting) or by volumetric UT examination (bolting). The VT-3 visual methods may be applied for the detection of cracking only when the flaw tolerance of the component or affected assembly, as evaluated for reduced fracture toughness properties, is known and has been shown to be tolerant of easily detected large flaws, even under reduced fracture toughness conditions. In addition, VT-3 examinations are used to monitor/inspect for loss of material induced by wear and for general aging conditions, such as gross distortion caused by void swelling and irradiation growth or by gross effects of loss of preload caused by thermal and irradiation-enhanced stress relaxation and creep.*
>
> *In addition, the program adopts the recommended guidance in MRP-227 for defining the Expansion criteria that need to be applied to inspections of Primary Components and Existing Requirement Components and for expanding the examinations to include additional Expansion Components. As a result, inspections performed on the RVI components are performed consistent with the inspection frequency and sampling bases for Primary Components, Existing Requirement Components, and Expansion Components in MRP-227, which have been demonstrated to be in conformance with the inspection criteria, sampling basis criteria, and sample Expansion criteria in Section A.1.2.3.4 of NRC Branch Position RLSB-1.*
>
> *Specifically, the program implements the parameters monitored/inspected criteria and bases for inspecting the relevant parameter conditions for Westinghouse designed Primary Components in Table 4-3 of MRP-227 and for Westinghouse designed Expansion Components in Table 4-6 of MRP-227.*
>
> *The program is supplemented by the following plant specific Primary Component and Expansion Component inspections for the program (as applicable): there are no supplemental Primary or Expansion components for the CNP program.*

> *In addition, in some cases (as defined in MRP-227), physical measurements are used as supplemental techniques to manage for the gross effects of wear, loss of preload due to stress relaxation, or for changes in dimension due to void swelling, deflection or distortion. The physical measurements methods applied in accordance with this program include the measurement of the 304 SS hold down springs in accordance with MRP-227 Table 4-3.*

The detection of aging effects credited for augmented inspections in the CNP RVI AMP are based on guidance in MRP-227-A and MRP-228. The program implements the guidance of MRP-227-A Table 4-3 and Table 4-6 which are included as Appendix A and Appendix B, respectively. This includes the measurement of the 304 SS hold down springs in each unit in accordance with MRP-227-A Table 4-3.

The CNP RVI AMP detection of aging effects is consistent with NUREG-1801, Revision 2, Section XI.M16A.

5.5 Monitoring and Trending

The following is NUREG-1801, Revision 2, Section XI.M16A, Evaluation and Technical Basis Element 5:

> *The methods for monitoring, recording, evaluating, and trending the data that result from the program's inspections are given in Section 6 of MRP-227 and its subsections. The evaluation methods include recommendations for flaw depth sizing and for crack growth determinations as well for performing applicable limit load, linear elastic and elastic plastic fracture analyses of relevant flaw indications. The examinations and re-examinations required by the MRP-227 guidance, together with the requirements specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel, provide timely detection, reporting, and corrective actions with respect to the effects of the age-related degradation mechanisms within the scope of the program. The extent of the examinations, beginning with the sample of susceptible PWR internals component locations identified as Primary Component locations, with the potential for inclusion of Expansion Component locations if the effects are greater than anticipated, plus the continuation of the Existing Programs activities, such as the ASME Code, Section XI, Examination Category B-N-3 examinations for core support structures, provides a high degree of confidence in the total program.*

The CNP RVI AMP implements MRP-227-A guidance for monitoring, recording, evaluating, and trending data that result from inspections. Inspections methodologies, inspection procedures, and inspection personnel guidance provided in MRP-228 will be followed. The program also credits monitoring performed by the ASME Section XI program.

The CNP RVI AMP monitoring and trending is consistent with NUREG-1801, Revision 2, Section XI.M16A.

5.6 Acceptance Criteria

The following is NUREG-1801, Revision 2, Section XI.M16A, Evaluation and Technical Basis Element 6:

> *Section 5 of MRP-227 provides specific examination acceptance criteria for the Primary and Expansion Component examinations. For components addressed by examinations referenced to ASME Code, Section XI, the IWB-3500 acceptance criteria apply. For other components covered by Existing Programs, the examination acceptance criteria are described within the Existing Program reference document.*
>
> *The guidance in MRP-227 contains three types of examination acceptance criteria:*
>
> - *For visual examination (and surface examination as an alternative to visual examination), the examination acceptance criterion is the absence of any of the specific, descriptive relevant conditions; in addition, there are requirements to record and disposition surface breaking indications that are detected and sized for length by VT-1/EVT-1 examinations;*
> - *For volumetric examination, the examination acceptance criterion is the capability for reliable detection of indications in bolting, as demonstrated in the examination Technical Justification; in addition, there are requirements for system-level assessment of bolted or pinned assemblies with unacceptable volumetric (UT) examination indications that exceed specified limits; and*
> - *For physical measurements, the examination acceptance criterion for the acceptable tolerance in the measured differential height from the top of the plenum rib pads to the vessel seating surface in B&W plants are given in Table 5-1 of MRP-227. The acceptance criterion for physical measurements performed on the height limits of the Westinghouse-designed hold-down springs will be developed prior to the first physical measurement.*

The CNP RVI AMP applies examination acceptance criteria provided in MRP-227-A, Section 5. In addition, WCAP-17096, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (WCAP-17096) has been developed by the PWROG and submitted by EPRI to the NRC for review and approval. I&M will evaluate degraded components using the supplemental guidance in WCAP-17096 as applicable. Application of guidance in WCAP-17096 will include any conditions or limitations resulting from the NRC review currently in-progress.

Acceptance criteria for the hold-down springs will be developed prior to the first physical measurement. Components inspected by the ASME Section XI program will be subject to acceptance criteria in the ASME Code as described in that program. The thimble tube inspection program contains acceptance criteria for inspection results.

The CNP RVI AMP acceptance criteria are consistent with NUREG-1801, Revision 2, Section XI.M16A.

5.7 Corrective Actions

The following is NUREG-1801, Revision 2, Section XI.M16A, Evaluation and Technical Basis Element 7:

> *Corrective actions following the detection of unacceptable conditions are fundamentally provided for in each plant's corrective action program. Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the plant corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection. The disposition will ensure that design basis functions of the reactor internals components will continue to be fulfilled for all licensing basis loads and events. Examples of methodologies that can be used to analytically disposition unacceptable conditions are found in the ASME Code, Section XI or in Section 6 of MRP-227. Section 6 of MRP-227 describes the options that are available for disposition of detected conditions that exceed the examination acceptance criteria of Section 5 of the report. These include engineering evaluation methods, as well as supplementary examinations to further characterize the detected condition, or the alternative of component repair and replacement procedures. The latter are subject to the requirements of the ASME Code, Section XI. The implementation of the guidance in MRP-227, plus the implementation of any ASME Code requirements, provides an acceptable level of aging management of safety-related components addressed in accordance with the corrective actions of 10 CFR Part 50, Appendix B or its equivalent, as applicable.*
>
> *Other alternative corrective action bases may be used to disposition relevant conditions if they have been previously approved or endorsed by the NRC. Examples of previously NRC-endorsed alternative corrective actions bases include those corrective actions bases for Westinghouse-design RVI components that are defined in Tables 4-1, 4-2, 4-3, 4-4, 4-5, 4-6, 4-7 and 4-8 of Westinghouse Report No. WCAP-14577-Rev. 1-A, or for B&W-designed RVI components in B&W Report No. BAW-2248. Westinghouse Report No. WCAP-14577-Rev. 1-A was endorsed for use in an NRC SE to the Westinghouse Owners Group, dated February 10, 2001. B&W Report No. BAW-2248 was endorsed for use in an SE to Framatome Technologies on behalf of the B&W Owners Group, dated December 9, 1999. Alternative corrective action bases not approved or endorsed by the NRC will be submitted for NRC approval prior to their implementation.*

Corrective actions are recorded and dispositioned in accordance with the CNP Corrective Actions Program (CAP) and Quality Assurance Program Description. The CAP includes procedure guidance for action initiation, condition action and closure, conduct of evaluations, conduct of effectiveness reviews, and conduct of causal evaluations.

The CNP RVI AMP corrective actions are consistent with NUREG-1801, Revision 2, Section XI.M16A.

5.8 Confirmation Process

The following is NUREG-1801, Revision 2, Section XI.M16A, Evaluation and Technical Basis Element 8:

> *Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B, or their equivalent, as applicable. It is expected that the implementation of the guidance in MRP-227 will provide an acceptable level of quality for inspection, flaw evaluation, and other elements of aging management of the PWR internals that are addressed in accordance with the 10 CFR Part 50, Appendix B, or their equivalent (as applicable), confirmation process, and administrative controls.*

The CNP Corrective Action Program and Quality Assurance Program Description have been developed in accordance with 10 CFR Part 50, Appendix B. The CNP RVI AMP implements the guidance in MRP-227-A.

The CNP RVI AMP confirmation process is consistent with NUREG-1801, Revision 2, Section XI.M16A.

5.9 Administrative Controls

The following is NUREG-1801, Revision 2, Section XI.M16A, Evaluation and Technical Basis Element 9:

> *The administrative controls for such programs, including their implementing procedures and review and approval processes, are under existing site 10 CFR 50 Appendix B Quality Assurance Programs, or their equivalent, as applicable. Such a program is thus expected to be established with a sufficient level of documentation and administrative controls to ensure effective long-term implementation.*

The CNP Quality Assurance Program Description has been developed in accordance with 10 CFR Part 50, Appendix B. The Quality Assurance Program Description ensures proper administrative controls on the CNP RVI AMP.

The CNP RVI AMP administrative controls are consistent with NUREG-1801, Revision 2, Section XI.M16A.

5.10 Operating Experience

The following is NUREG-1801, Revision 2, Section XI.M16A, Evaluation and Technical Basis Element 10:

Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. A summary of observations to date is provided in Appendix A of MRP-227-A. The applicant is expected to review subsequent operating experience for impact on its program or to participate in industry initiatives that perform this function.

The application of the MRP-227 guidance will establish a considerable amount of operating experience over the next few years. Section 7 of MRP-227 describes the reporting requirements for these applications, and the plan for evaluating the accumulated additional operating experience.

I&M will continue to be engaged with industry groups for sharing and reviewing OE in accordance with the Operating Experience Program. In addition, I&M will maintain industry involvement as described in Section 3.4. I&M will follow reporting requirements provided in MRP-227-A.

The CNP RVI AMP operating experience is consistent with in NUREG-1801, Revision 2, Section XI.M16A.

6.0 REFERENCES

6.1 American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2004 Edition, American Society of Mechanical Engineers, New York, NY.
6.2 CNP Document 01-DCP-0125, "Reactor Vessel Core Barrel – Replace Missing Bolts at Locations A-4, A-5 and A-6," March 1997
6.3 CNP Document 1-MOD-55520, "Replace Unit 1 Reactor Vessel Closure Head (1-OME-1)," July 2006.
6.4 CNP Document 2-MOD-55516, "Replace Unit 2 Reactor Vessel Closure Head (2-OME-1)," June 2007.
6.5 CNP Document 2-OHP-SP-045, "Unit 2 Cycle V-VI Refueling Procedure," May 1986
6.6 CNP Document 12-EHP-6040-PER-324, Revision 6, "Incore Instrumentation Thimble Tube Multifrequency Eddy Current Inspection," December 2011.
6.7 CNP Document "Cook Nuclear Plant Primary Strategic Water Chemistry Plan," Revision 7, November 2011.
6.8 CNP Document AEP-NRC-2011-38, "Revision to Regulatory Commitments Associated with Application for Renewed Operating Licenses," September 2011.
6.9 CNP Document AR 2010-1804, "Rx Vessel Core Support Lug Bolting Anomalies," Originated March 2010.
6.10 CNP Document AR 2010-10940, "Debris Found in 1-OME-1 on the Core Plate," Originated October 2010.
6.11 CNP Document Contract-6223, "Control Rod Guide Tube Support Pin Replacement," June 1985.
6.12 CNP Document "D. C. Cook Nuclear Plant Updated Final Safety Analysis Report," Revision 24, March 2012.
6.13 CNP Document EC-0000050972, Revision 2, "Replace Reactor Vessel Baffle Bolts," November 2010.
6.14 CNP Document EC-0000051640, "RX Vessel Lower Radial Support System (LRSS) Clevis Replacement Bolting for Unit 1," July 2012.
6.15 CNP Document GT 00846697, "License Renewal Implementation (Y10) for the RVI Program," Originated February 2009.
6.16 CNP Document GT 2012-1808, "Unit 1 CRGT Split Pin Replacement," Originated February 2012.
6.17 CNP Document GT 2012-1809, "Unit 2 CRGT Split Pin Replacement," Originated February 2012.
6.18 CNP Document ISI PROGRAM 4TH INTERVAL, Revision 2, "ISI Program Plan Fourth Ten-Year Inservice Inspection interval Donald C. Cook Nuclear Plant, Units 1 & 2," May 2011.
6.19 CNP Document "License Renewal Application: Donald C. Cook Nuclear Plant," October 2003.
6.20 CNP Document LRP-EAMP-01, Revision 3, "Evaluation of Aging Management Programs for License Renewal," November 2005.
6.21 CNP Document PMI-7030, Revision 40, "Corrective Action Program," May 2012.

6.22 CNP Document QAPD, Revision 22, "Donald C. Cook Nuclear Plant Quality Assurance Program Description," May 2012.

6.23 CNP Document RFC-01-2858, "Work to support Control Rod Guide Tube Replacement," June 1985.

6.24 CNP Document RFC-DC-01-2353, "Removal of Eight Part Length Rods, Installation of Anti-Rotation Devices on Each Part Length Rod CRDM, Install Eight Thimble Plug Devices in Place of the Part Length Rod," November 1978.

6.25 CNP Document RFC-DC-02-988, "Modify Unit 2 Fuel Assemblies from a 15X15 to a 17X17 Fuel Rod Array," June 1976.

6.26 CNP Document RFC-DC-02-2355, "Installation of Permanent Anti-Rotational Devices for Part Length CRDM Lead Screws," December 1978.

6.27 CNP Document RFC-DC-02-2924, "Control Rod Guide Tube Cap Screw Modifications," May 1986

6.28 *Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191)*. EPRI, Palo Alto, CA: 2006. 1013234.

6.29 *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)*. EPRI, Palo Alto, CA: 2011. 1022863.

6.30 *Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228)*. EPRI, Palo Alto, CA: 2009. 1016609.

6.31 NEI Document NEI 03-08, Revision 2, "Guideline for the Management of Materials Issues," Nuclear Energy Institute, Washington, DC, January 2010.

6.32 NRC Document ML11308A770, "Revision 1 to the Final Safety Evaluation of the Electric Power Research institute (EPRI) Report, Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, 'Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines' (TAC NO. ME0680," December 2011.

6.33 NRC Document NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," December 2010.

6.34 NRC Document NUREG-1831, "Safety Evaluation Report Related to the License Renewal of the Donald C. Cook Nuclear Plant, Units 1 and 2," Docket Nos. 50-315 and 50-316. Indiana Michigan Power Company, July 2005.

6.35 NRC Document NRCB 88-09, "Thimble Tube Thinning in Westinghouse Reactors," July 1988.

6.36 NRC Document RIS 11-07 "License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management," July 2011.

6.37 Westinghouse Document "Attachment: Description of Additional Guide Tube Repairs for D.C. Cook No. 1"

6.38 Westinghouse Document WCAP-11000, "D. C. Cook Unit 2 Estimated Operability with Failed Control Rod Guide Tube Support Pins," January 1985.

6.39 Westinghouse Document WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals," March, 2001.

6.40 Westinghouse Document WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," December 2009

6.41 Westinghouse Document WCAP-17300, Revision 0, "Reactor Vessel Internals Program Plan for Aging Management of Reactor Internals at D.C. Cook Nuclear Plant Unit 1," February 2011.

6.42 Westinghouse Document WCAP-17301, Revision 0, "Reactor Vessel Internals Program Plan for Aging Management of Reactor Internals at D.C. Cook Nuclear Plant Unit 2," February 2011.

APPENDIX A: PRIMARY INSPECTION COMPONENTS

The following is Table 4-3 "Westinghouse plants Primary components" from MRP-227-A. The CNP RVI AMP implements the guidance provided in this table.

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Loss of Material (Wear)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period, and no earlier than two refueling outages prior to the start of the license renewal period. Subsequent examinations are required on a ten-year interval.	20% examination of the number of CRGT assemblies, with all guide cards within each selected CRGT assembly examined. See Figure 4-20
Control Rod Guide Tube Assembly Lower flange welds	All plants	Cracking (SCC, Fatigue) Aging Management (IE and TE)	Bottom-mounted instrumentation (BMI) column bodies, Lower support column bodies (cast) Upper core plate Lower support forging/casting	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal on the individual periphery CRGT assemblies. (Note 2) See Figure 4-21
Core Barrel Assembly Upper core barrel flange weld	All plants	Cracking (SCC)	Lower support column bodies (non cast) Core barrel outlet nozzle welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4). See Figure 4-22
Core Barrel Assembly Upper and lower core barrel cylinder girth welds	All plants	Cracking (SCC, IASCC, Fatigue)	Upper and lower core barrel cylinder axial welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4). See Figure 4-22

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Barrel Assembly Lower core barrel flange weld (Note 5)	All plants	Cracking (SCC, Fatigue)	None	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4). See Figure 4-22
Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts	Cracking (IASCC, Fatigue) that results in • Lost or broken locking devices • Failed or missing bolts • Protrusion of bolt heads Aging Management (IE and ISR) (Note 6)	None	Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Bolts and locking devices on high fluence seams. 100% of components accessible from core side (Note 3). See Figure 4-23
Baffle-Former Assembly Baffle-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR) (Note 6)	Lower support column bolts, Barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval.	100% of accessible bolts (Note 3). Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See Figures 4-23 and 4-24

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Baffle-Former Assembly Assembly (Includes: Baffle plates, baffle edge bolts and indirect effects of void swelling in former plates)	All plants	Distortion (Void Swelling), or Cracking (IASCC) that results in • Abnormal interaction with fuel assemblies • Gaps along high fluence baffle joint • Vertical displacement of baffle plates near high fluence joint • Broken or damaged edge bolt locking systems along high fluence baffle joint	None	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Core side surface as indicated. See Figures 4-24, 4-25, 4-26 and 4-27
Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs	Distortion (Loss of Load) Note: This mechanism was not strictly identified in the original list of age-related degradation mechanisms [7].	None	Direct measurement of spring height within three cycles of the beginning of the license renewal period. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years.	Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to minimize uncertainty. See Figure 4-28

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields	Cracking (Fatigue) or Loss of Material (Wear) that results in thermal shield flexures excessive wear, fracture, or complete separation	None	Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of thermal shield flexures. See Figures 4-29 and 4-36

Notes to Table 4-3:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.
2. A minimum of 75% of the total identified sample population must be examined.
3. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined for inspection credit.
4. A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined from either the inner or outer diameter for inspection credit.
5. The lower core barrel flange weld may be alternatively designated as the core barrel-to-support plate weld in some Westinghouse plant designs.
6. Void swelling effects on this component is managed through management of void swelling on the entire baffle-former assembly.

APPENDIX B: EXPANSION INSPECTION COMPONENTS

The following is Table 4-6 “Westinghouse plants Expansion components” from MRP-227-A. The CNP RVI AMP implements the guidance provided in this table.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Upper Internals Assembly Upper core plate	All plants	Cracking (Fatigue, Wear)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2).
Lower Internals Assembly Lower support forging or castings	All plants	Cracking Aging Management (TE in Casting)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2). See Figure 4-33.
Core Barrel Assembly Barrel-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE, Void Swelling and ISR)	Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% of accessible bolts. Accessibility may be limited by presence of thermal shields or neutron pads (Note 2). See Figure 4-23.
Lower Support Assembly Lower support column bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% of accessible bolts or as supported by plant-specific justification (Note 2). See Figures 4-32 and 4-33.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Barrel Assembly Core barrel outlet nozzle welds	All plants	Cracking (SCC, Fatigue) Aging Management (IE of lower sections)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 2) See Figure 4-22
Core Barrel Assembly Upper and lower core barrel cylinder axial welds	All plants	Cracking (SCC, IASCC) Aging Management (IE)	Upper and lower core barrel cylinder girth welds	Enhanced visual (EVT-1) examination. e-inspection every 10 years following initial inspection.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 2). See Figure 4-22
Lower Support Assembly Lower support column bodies (non cast)	All plants	Cracking (IASCC) Aging Management (IE)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2). See Figure 4-34.
Lower Support Assembly Lower support column bodies (cast)	All plants	Cracking (IASCC) including the detection of fractured support columns Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible support columns (Note 2). See Figure 4-34.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Bottom Mounted Instrumentation System Bottom-mounted instrumentation (BMI) column bodies	All plants	Cracking (Fatigue) including the detection of completely fractured column bodies Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles. Re-inspection every 10 years following initial inspection. Flux thimble insertion/withdrawal to be monitored at each inspection interval.	100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal. See Figure 4-35.

Notes to Table 4-6:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.
2. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).

APPENDIX C: EXISTING PROGRAMS COMPONENTS

The following is Table 4-9 "Westinghouse plants Existing Programs components" from MRP-227-A. The CNP RVI AMP is consistent with the information provided in this table.

Item	Applicability	Effect (Mechanism)	Reference	Examination Method	Examination Coverage
Core Barrel Assembly Core barrel flange	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination to determine general condition for excessive wear.	All accessible surfaces at specified frequency.
Upper Internals Assembly Upper support ring or skirt	All plants	Cracking (SCC, Fatigue)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
Lower Internals Assembly Lower core plate XL lower core plate (Note 1)	All plants	Cracking (IASCC, Fatigue) Aging Management (IE)	ASME Code Section XI	Visual (VT-3) examination of the lower core plates to detect evidence of distortion and/or loss of bolt integrity.	All accessible surfaces at specified frequency.
Lower Internals Assembly Lower core plate XL lower core plate (Note 1)	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
Bottom Mounted Instrumentation System Flux thimble tubes	All plants	Loss of material (Wear)	NUREG-1801 Rev. 1	Surface (ET) examination.	Eddy current surface examination as defined in plant response to IEB 88-09.
Alignment and Interfacing Components Clevis insert bolts	All plants	Loss of material (Wear) (Note 2)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
Alignment and Interfacing Components Upper core plate alignment pins	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.

Notes to Table 4-9:

1. XL = "Extra Long" referring to Westinghouse plants with 14-foot cores.
2. Bolt was screened in because of stress relaxation and associated cracking; however, wear of the clevis/insert is the issue.

APPENDIX D: ACCEPTANCE AND EXPANSION CRITERIA

The following is Table 5-3 "Westinghouse plants examination acceptance and expansion criteria" from MRP-227-A. The CNP RVI AMP implements the guidance provided in this table.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Visual (VT-3) examination. The specific relevant condition is wear that could lead to loss of control rod alignment and impede control assembly insertion.	None	N/A	N/A
Control Rod Guide Tube Assembly Lower flange welds	All plants	Enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	a. Bottom-mounted instrumentation (BMI) column bodies b. Lower support column bodies (cast), upper core plate and lower support forging or casting	a. Confirmation of surface-breaking indications in two or more CRGT lower flange welds, combined with flux thimble insertion/withdrawal difficulty, shall require visual (VT-3) examination of BMI column bodies by the completion of the next refueling outage. b. Confirmation of surface-breaking indications in two or more CRGT lower flange welds shall require EVT-1 examination of cast lower support column bodies, upper core plate and lower support forging/castings within three fuel cycles following the initial observation.	a. For BMI column bodies, the specific relevant condition for the VT-3 examination is completely fractured column bodies. b. For cast lower support column bodies, upper core plate and lower support forging/castings, the specific relevant condition is a detectable crack-like surface indication.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Upper core barrel flange weld	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	a. Core barrel outlet nozzle welds b. Lower support column bodies (non cast)	a. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel flange weld shall require that the EVT-1 examination be expanded to include the core barrel outlet nozzle welds by the completion of the next refueling outage. b. If extensive cracking in the core barrel outlet nozzle welds is detected, EVT-1 examination shall be expanded to include the upper six inches of the accessible surfaces of the non-cast lower support column bodies within three fuel cycles following the initial observation.	a and b. The specific relevant condition for the expansion core barrel outlet nozzle weld and lower support column body examination is a detectable crack-like surface indication.
Core Barrel Assembly Lower core barrel flange weld (Note 2)	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	None	None

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Upper core barrel cylinder girth welds	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Upper core barrel cylinder axial welds	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the upper core barrel cylinder axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion upper core barrel cylinder axial weld examination is a detectable crack-like surface indication.
Core Barrel Assembly Lower core barrel cylinder girth welds	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Lower core barrel cylinder axial welds	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the lower core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the lower core barrel cylinder axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion lower core barrel cylinder axial weld examination is a detectable crack-like surface indication.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts	Visual (VT-3) examination. The specific relevant conditions are missing or broken locking devices, failed or missing bolts, and protrusion of bolt heads.	None	N/A	N/A
Baffle-Former Assembly Baffle-former bolts	All plants	Volumetric (UT) examination. The examination acceptance criteria for the UT of the baffle-former bolts shall be established as part of the examination technical justification.	a. Lower support column bolts b. Barrel-former bolts	a. Confirmation that more than 5% of the baffle-former bolts actually examined on the four baffle plates at the largest distance from the core (presumed to be the lowest dose locations) contain unacceptable indications shall require UT examination of the lower support column bolts within the next three fuel cycles. b. Confirmation that more than 5% of the lower support column bolts actually examined contain unacceptable indications shall require UT examination of the barrel-former bolts.	a and b. The examination acceptance criteria for the UT of the lower support column bolts and the barrel-former bolts shall be established as part of the examination technical justification.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Baffle-Former Assembly Assembly	All plants	Visual (VT-3) examination. The specific relevant conditions are evidence of abnormal interaction with fuel assemblies, gaps along high fluence shroud plate joints, vertical displacement of shroud plates near high fluence joints, and broken or damaged edge bolt locking systems along high fluence baffle plate joints.	None	N/A	N/A
Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs	Direct physical measurement of spring height. The examination acceptance criterion for this measurement is that the remaining compressible height of the spring shall provide hold-down forces within the plant-specific design tolerance.	None	N/A	N/A

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields	Visual (VT-3) examination. The specific relevant conditions for thermal shield flexures are excessive wear, fracture, or complete separation.	None	N/A	N/A

Notes to Table 5-3:
1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).
2. The lower core barrel flange weld may alternatively be designated as the core barrel-to-support plate weld in some Westinghouse plant designs.

APPENDIX E: REACTOR COMPONENT ILLUSTRATIONS

The following reactor component figures have been reproduced from MRP-227-A.

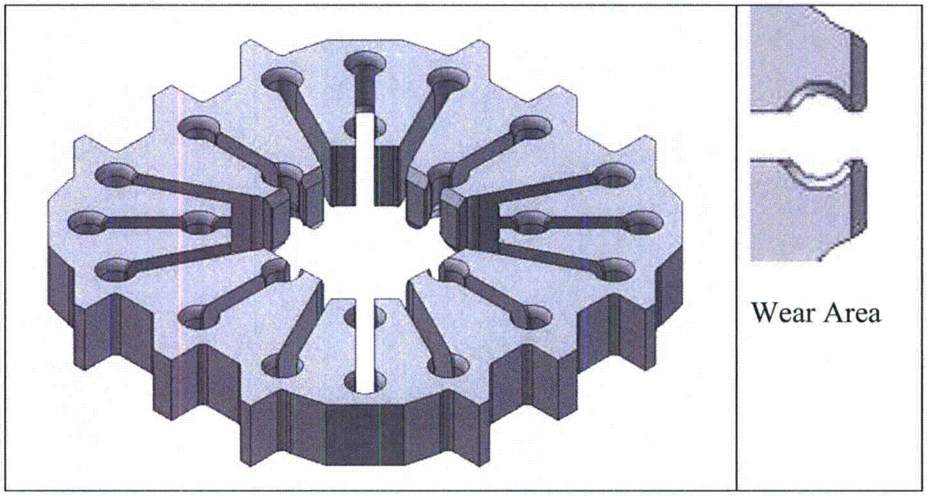

Figure E-1 Typical Westinghouse Control Rod Guide Card

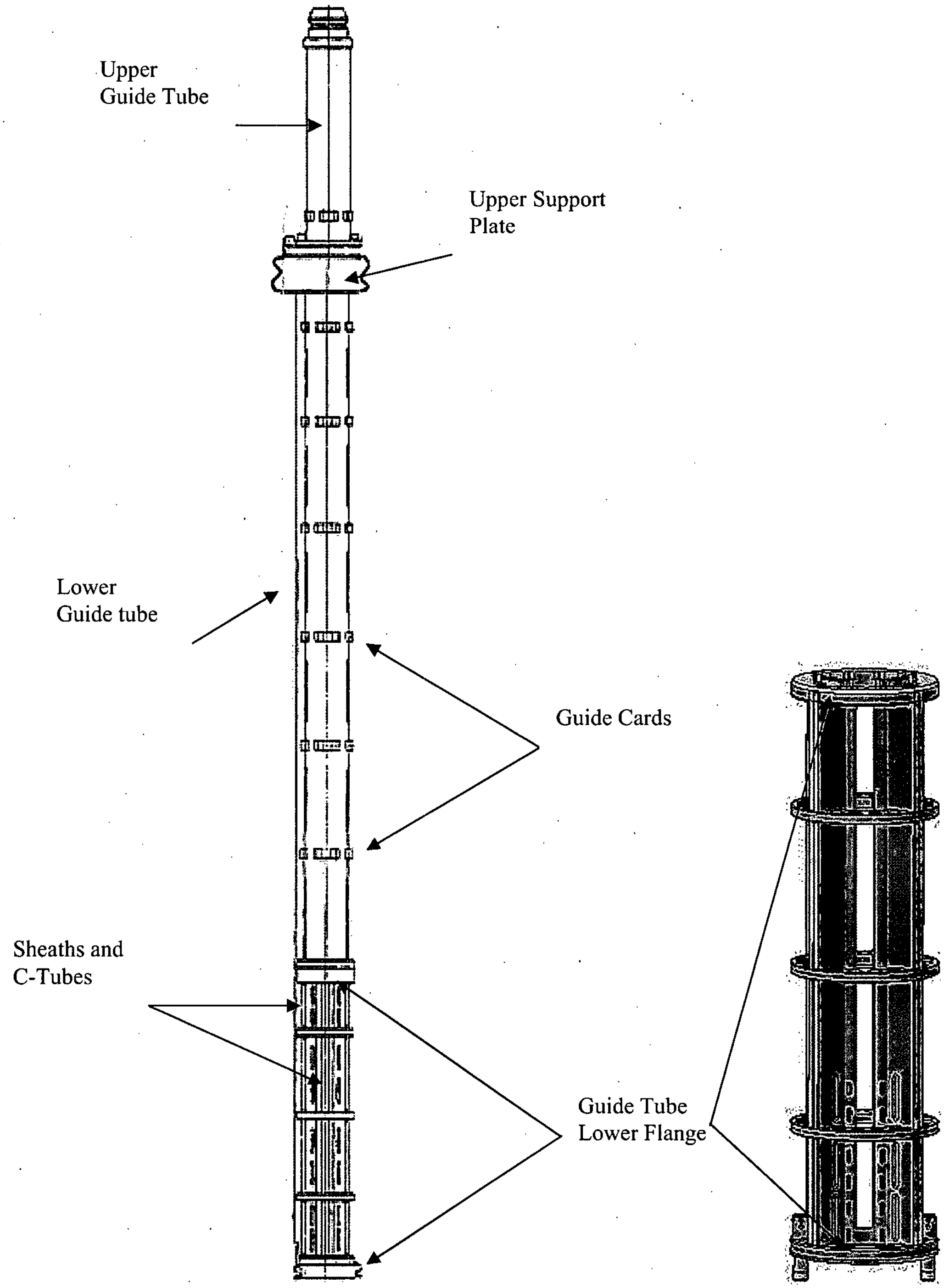

Figure E-2 Lower Section of Control Rod Guide Tube Assembly

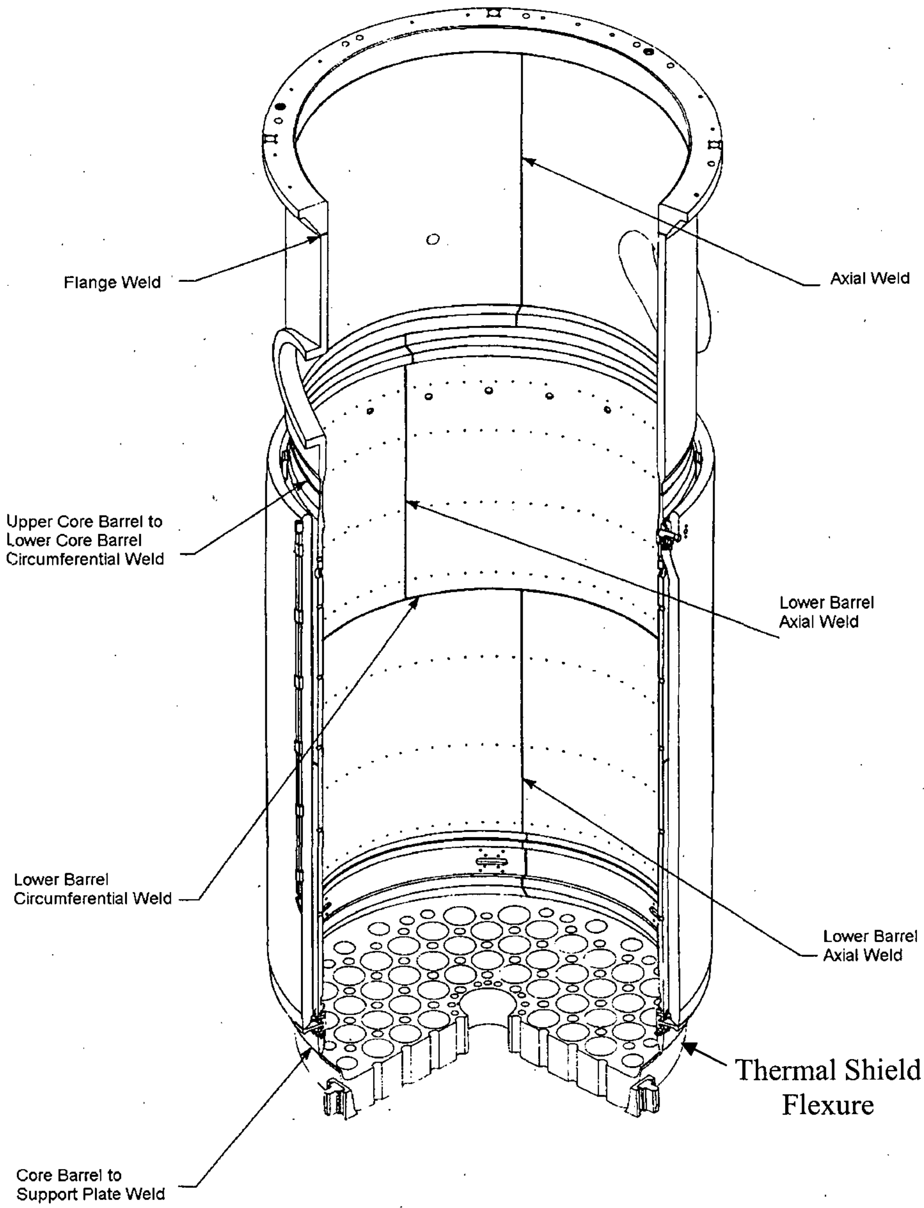

Figure E-3 Major Core Barrel Welds

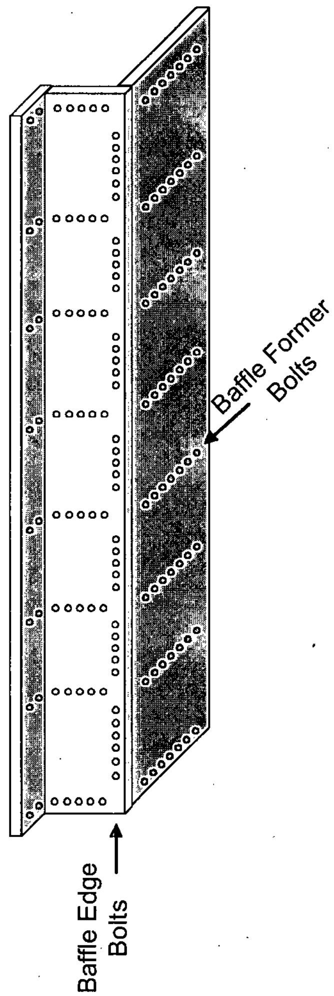

Figure E-4 Bolting Systems Used in Westinghouse Core Baffles

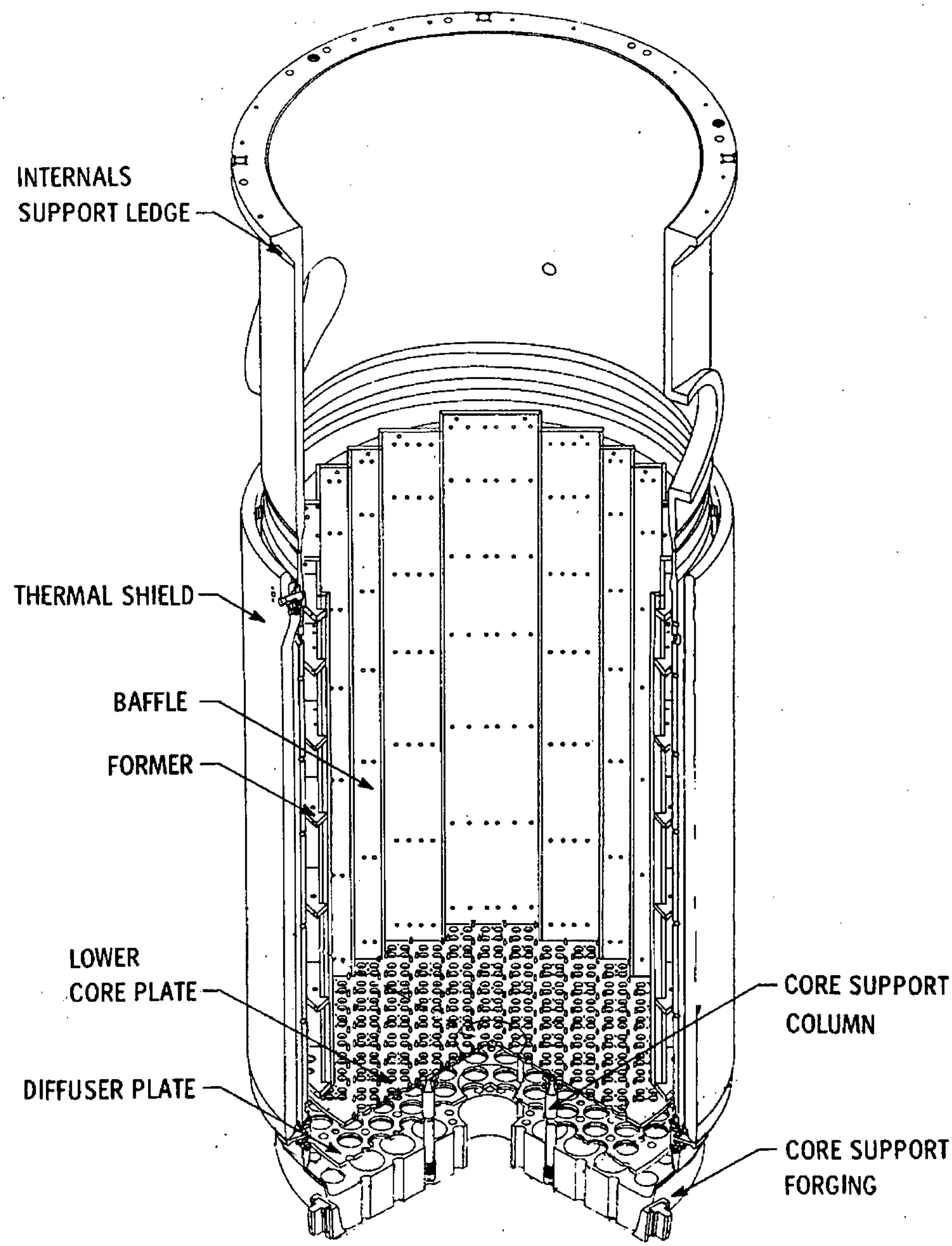

Figure E-5 Core Baffle/Barrel Structure

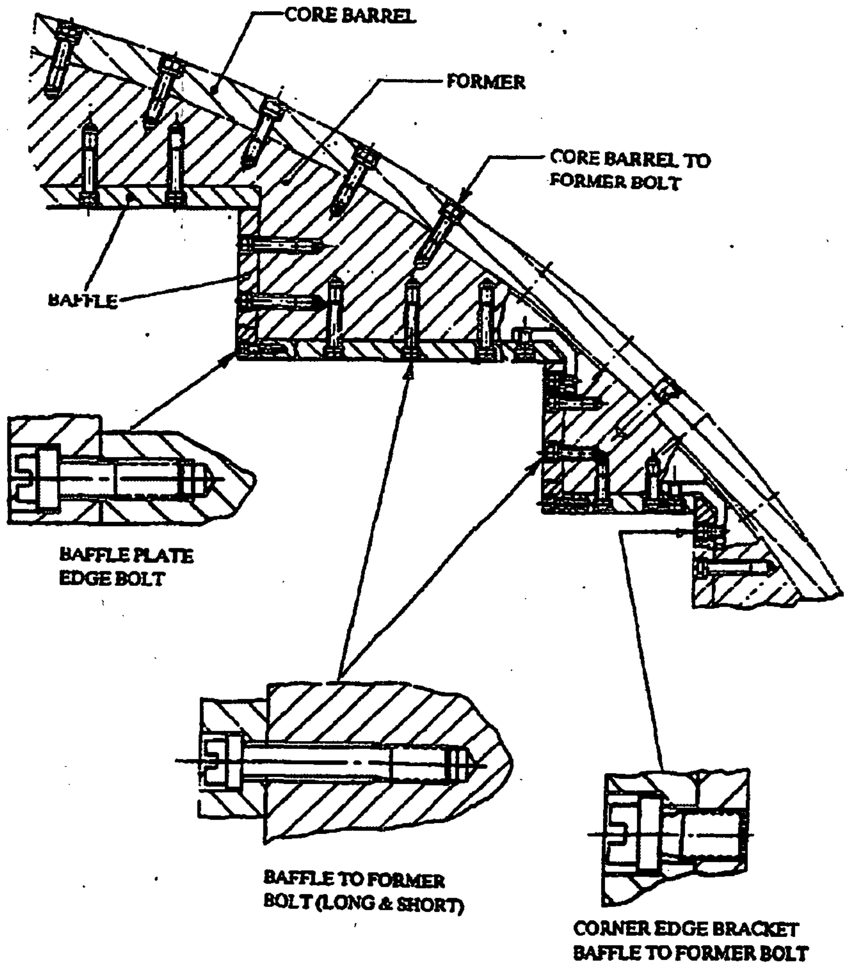

Figure E-6 Bolting in a Typical Westinghouse Baffle-Former Structure

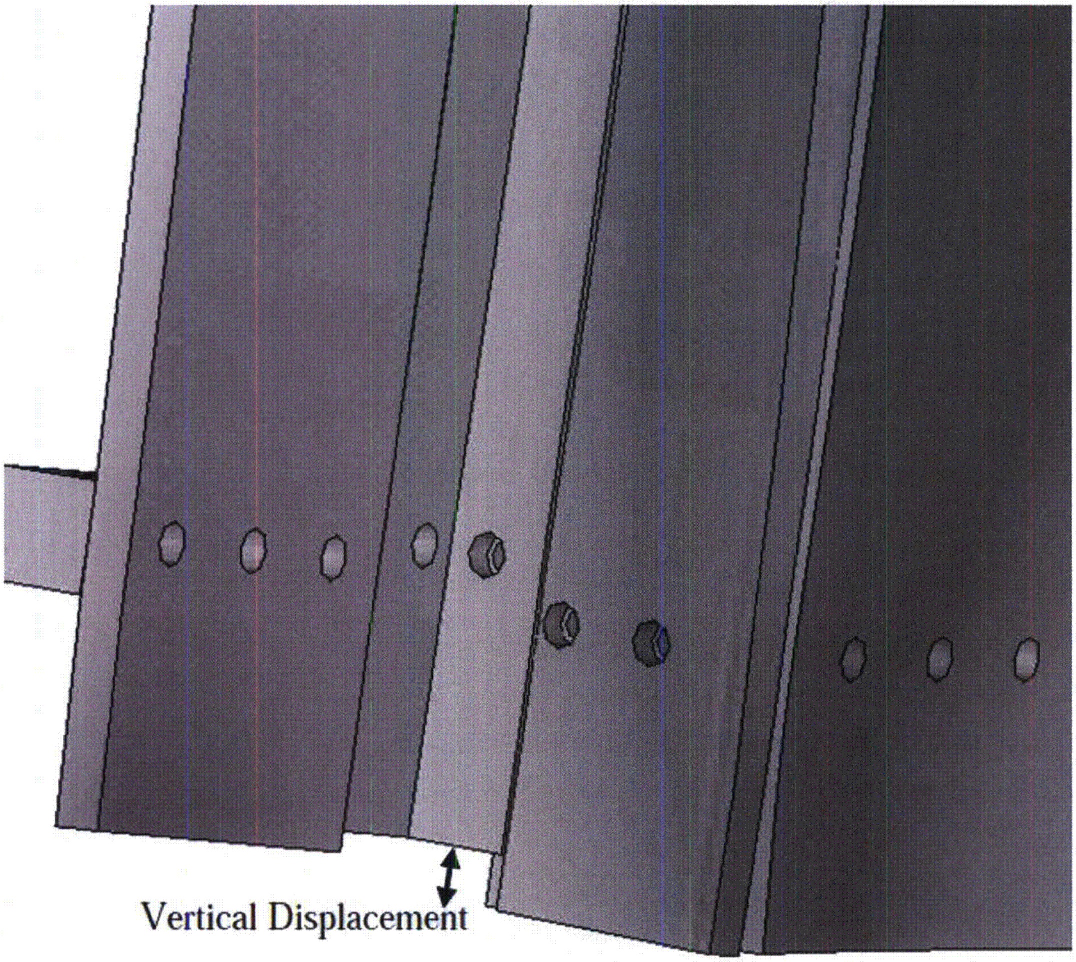

Figure E-7 Vertical Displacement between the Baffle Plates and Bracket at the Bottom of the Baffle-Former-Barrel Assembly (exaggerated)

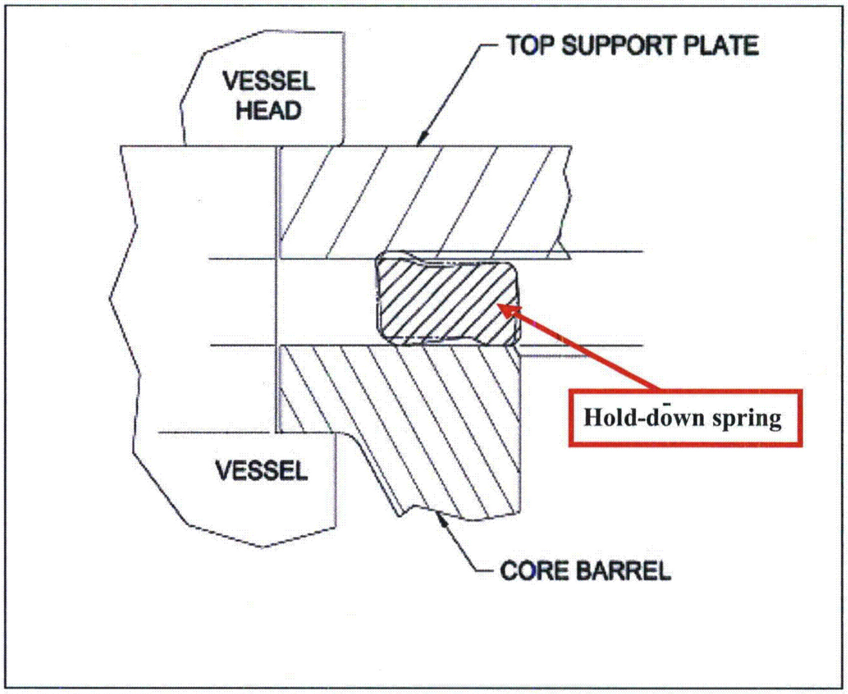

Figure E-8 Schematic Cross-Sections of the Westinghouse Hold-down Springs

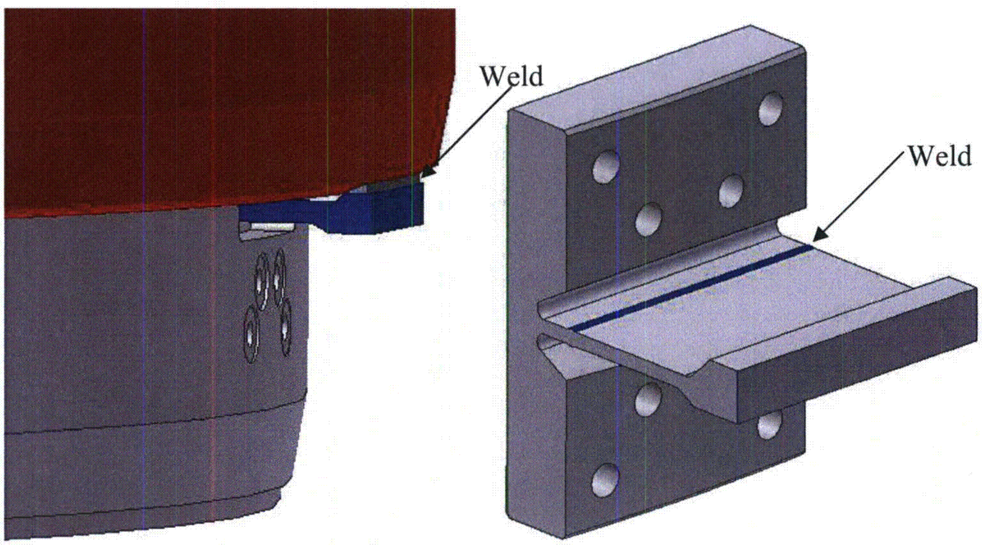

Figure E-9 Typical Thermal Shield Flexure

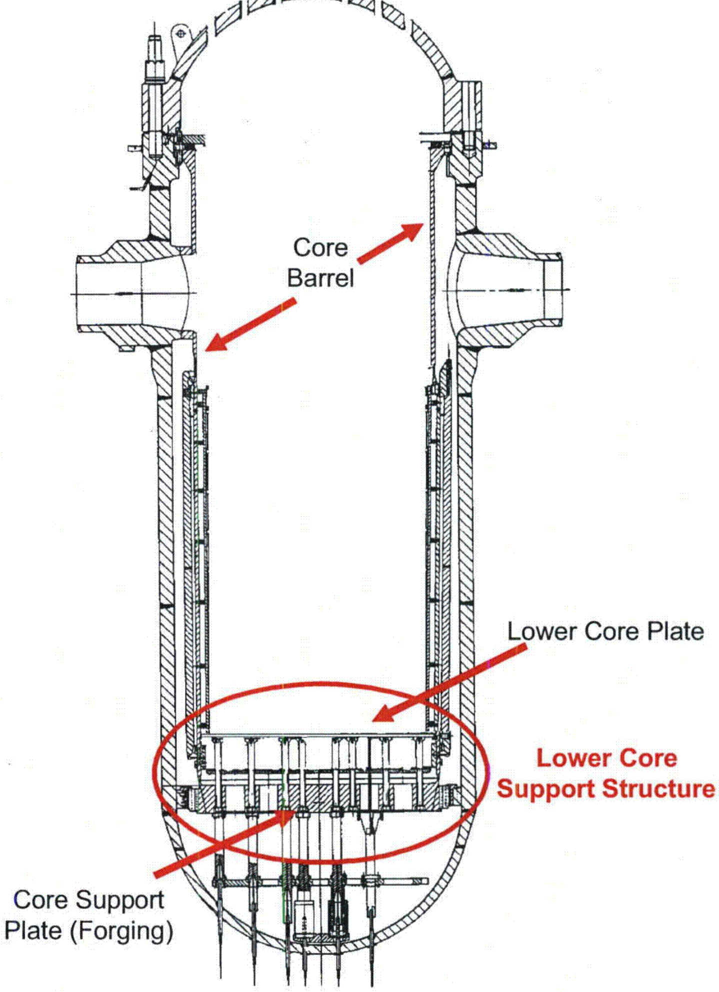

Figure E-10 Lower Core Support Structure

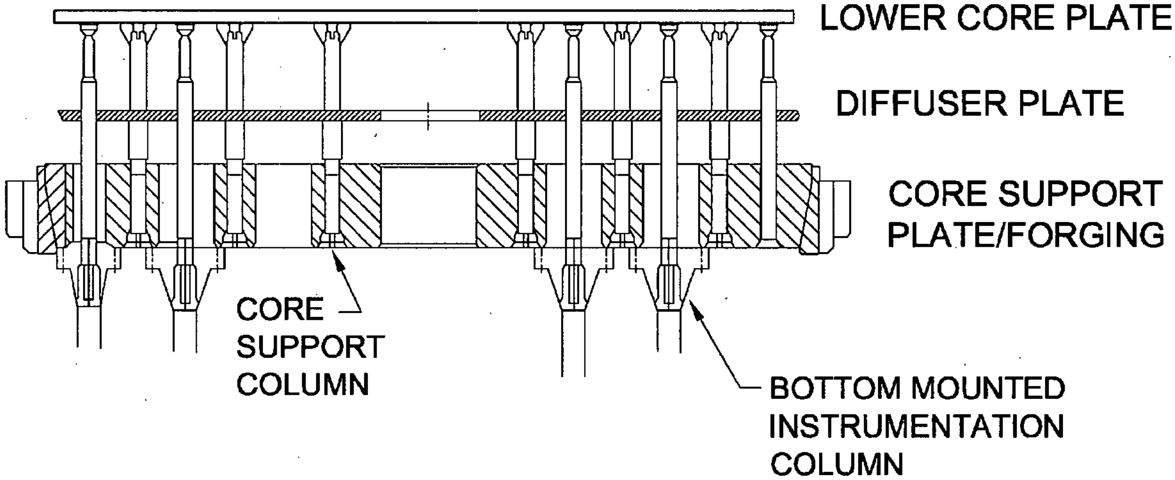

Figure E-11 Lower Core Support Structure – Core Support Plate Cross-Section

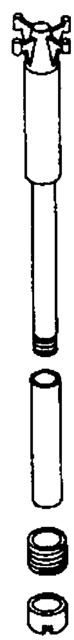

Figure E-12 Typical Core Support Column

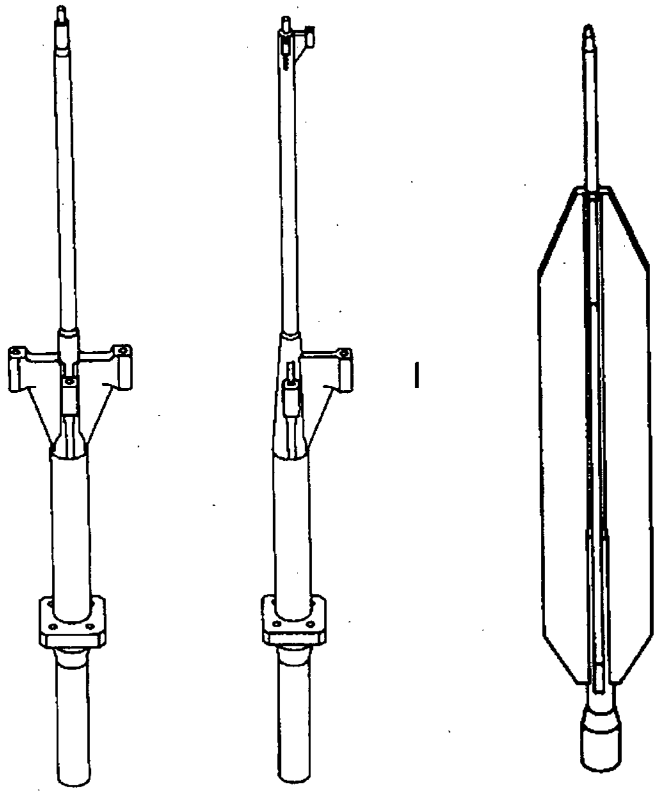

Figure E-13 Examples of Bottom-Mounted Instrumentation (BMI) Column Designs

ENCLOSURE 2 TO AEP-NRC-2012-82

DONALD C. COOK NUCLEAR PLANT
LIST OF REGULATORY COMMITMENTS

List of Regulatory Commitments

The following table identifies those actions committed to by Indiana Michigan Power Company in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	SCHEDULED COMPLETION DATE
This Action Item addresses the applicability of the FMECA and functionality analysis assumptions made in the development of MRP-227-A to individual facilities. I&M is participating in PWROG project PA-MSC-0938, "Support for Applicant Action Items 1, 2, and 7 from the Final Safety Evaluation on MRP-227, Revision 0" to address this item. The results of the evaluation will be provided to the NRC prior to the period of extended operation for each unit.	Unit 1: October 25, 2014 Unit 2: December 23, 2017
This Action Item requires the licensee to verify that all the RVI components within the scope for license renewal at that facility have been considered in applicable documents in development of MRP-227-A. I&M is participating in PWROG project PA-MSC-0938, "Support for Applicant Action Items 1, 2, and 7 from the Final Safety Evaluation on MRP-227, Revision 0" to address this item. The results of the evaluation will be provided to the NRC prior to the period of extended operation for each unit.	Unit 1: October 25, 2014 Unit 2: December 23, 2017
CNP Unit 1 and Unit 2 both have X-750 split pins. Project requests have been initiated to investigate split pin replacement for each unit. I&M will provide the NRC with the strategy for managing split pins prior to the period of extended operation for each unit.	Unit 1: October 25, 2014 Unit 2: December 23, 2017
CNP Unit 1 and Unit 2 both have 304 SS hold down springs. MRP-227-A guidance includes physical measurement of 304 SS hold down springs. This action item requires acceptance criteria to be provided to the NRC. CNP plant specific acceptance criteria will be developed and submitted to the NRC prior to the first required physical measurement. The hold down springs will be replaced if acceptance criteria are not developed in lieu of performing the first required physical measurement.	Unit 1: Prior to the first required physical measurement. Unit 2: Prior to the first required physical measurement.

A plant specific evaluation of RVI CASS materials is required in this Action Item. I&M is participating in PWROG project PA-MSC-0938, "Support for Applicant Action Items 1, 2, and 7 from the Final Safety Evaluation on MRP-227, Revision 0" to address this item The results of the evaluation will be provided to the NRC prior to the period of extended operation for each unit.	Unit 1: October 25, 2014 Unit 2: December 23, 2017